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Vol. 2

PRA PROCEDURES GUIDE

A Guide to the Performance of Probabilistic
Risk Assessments for Nuclear Power Plants

Final Report
Vol. 1 - Chapters 1-8
Vol. 2 - Chapters 9-13 and Appendices A-G

Prepared under the auspices of
The American Nuclear Society and
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Foreword

The development of safety design requirements for nuclear power plants in the last 20 to 25 years took place in a subjective, deterministic framework. Little use was made of the techniques of quantitative probabilistic risk assessment (PRA), largely because these techniques were not fully developed for analyzing nuclear power plants. It was F. R. Farmer who introduced the idea of reactor safety based on the reliability of consequence-limiting equipment in the early 1960s. The first major application of PRA techniques was the Reactor Safety Study (WASH-1400), which demonstrated that a nuclear power plant could be analyzed in a systematic fashion by PRA techniques. Since the completion of the Study in 1975, the Nuclear Regulatory Commission (NRC) has been exploring ways of systematically applying probabilistic analysis to nuclear power plants, and the use of PRA techniques has been rapidly becoming more widespread in the nuclear community.

Contributing to these developments has been a growing appreciation of the wisdom of the strong recommendations made by the Lewis Committee to use PRA techniques for reexamining the fabric of NRC's regulatory processes to make them more rational.* After the accident at Three Mile Island, these recommendations were reinforced by the Kemeny† and Rogovin reports,‡ which also encouraged the use of these techniques. As Lewis stated in his March 1981 Scientific American article,§ "the Three Mile Island incident illustrates graphically how important it is to quantify both the probability and the consequences of an accident, and to generate some public awareness of these issues.... This is an issue that goes to the heart of many regulatory and safety decisions, where one must have some measure of the risks one is willing to accept on as quantitative a basis as the expert community can provide."

The NRC has recently raised questions about potential accident risks for nuclear plants near high population concentrations. To answer these questions, the industry has performed PRAs for the Indian Point, Limerick, and Zion plants. Moreover, the utilities themselves are showing considerable interest in taking advantage of the safety and availability insights afforded by risk assessments. As a result of these forces, an increasing number of PRAs are either under way or being planned. Finally, the NRC is contemplating a future program (National Reliability Evaluation Program, NREP) in which many licensed nuclear power plants will be required to perform a probabilistic risk assessment.

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*H. W. Lewis et al., Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission, USNRC Report NUREG/CR-0400, 1978.

†J. G. Kemeny et al., Report of the President's Commission on the Accident at Three Mile Island, Pergamon Press, 1979.

‡M. Rogovin, Three Mile Island: A Report to the Commissioners and to the Public, USNRC Report NUREG/CR-1250 (Vol. 1), 1979.

§H. W. Lewis, "The Safety of Fission Reactors," Scientific American, March 1981.

Because of this increasing application of PRA techniques within the industry and the regulatory process, there is a need for technical guidance on methods and procedures. It was this need that led to the creation of the PRA Procedures Guide project and ultimately to this document.

The objective of this project was to compile a procedures guide describing the principal methods now used in PRAs. To accomplish these objectives, a Steering Committee and a Technical Writing Group were formed. Funding has been provided by the NRC, the Department of Energy (DOE), and the Electric Power Research Institute (EPRI), and expertise was contributed by the nuclear industry.

The group responsible for the document is the Steering Committee. The Committee includes representatives from the American Nuclear Society, the Institute of Electrical and Electronics Engineers, the NRC, the DOE, the Atomic Industrial Forum, EPRI, and utilities (see Chapter 1 and Appendix B for the membership list). The Technical Writing Group, whose members were selected by the Steering Committee (see Appendix B), consists of technical specialists experienced in the application of probabilistic and reliability techniques to the analysis of nuclear power plants.

To obtain the wide peer review desired for the Procedures Guide, the Steering Committee decided on two mechanisms: criticism by a carefully selected peer review group and open review in two conferences. The objective in establishing the peer review group was to bring additional technical expertise and, in some instances, alternative viewpoints to the project. An effort was also made to include experts who are not members of the nuclear community. Candidates for the peer group were proposed by the Steering Committee and members of the Technical Writing Group; those who were finally selected are listed in Appendix B.

The first of the two conferences, held on October 26-28, 1981, included a series of workshops in risk assessment. It was sponsored by the Institute of Electrical and Electronics Engineers. The second was held on April 4-7, 1982, by the American Nuclear Society. These meetings have allowed the Steering Committee to obtain comments from a large number of experts in disciplines related to probabilistic risk assessment as well as potential users of the Procedures Guide. The disposition of these comments, like those of the peer review group, has been resolved by the Technical Writing Group under the guidance of the Steering Committee.

Actual writing of the Procedures Guide by the Technical Writing Group began only in April 1981, and by July a working draft was produced for review by the Steering Committee. It was followed by a review draft that was distributed for peer review and discussion at the October 1981 conference. The October 1981 conference was heavily attended, and many comments were submitted to the Steering Committee. A major revision of the Procedures Guide resulted in a second draft, published in April 1982 for the attendees of the ANS Executive Conference, which reflected many, but not all, of the comments.

After the ANS Executive Conference, a final revision was made, and this document resulted. Thus, the methods described herein have received broad review from both PRA practitioners and potential users of PRA techniques.

Upon completion of the PRA Procedures Guide project, the Steering Committee, which has guided the project, was disbanded. Future questions or comments on the Guide should be directed to Robert M. Bernero, Division of Risk Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

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VOLUME 2

Chapter 9

Environmental Transport and Consequence Analysis

9.1 INTRODUCTION

9.1.1 OBJECTIVE AND SCOPE

This chapter describes how to calculate the consequences of radionuclide releases into the environment and how to interpret the results of such calculations. It is primarily intended for the would-be user of consequence models, someone who has perhaps obtained a consequence-modeling computer package off the shelf and wishes to know what to do with it--the information that is required as input, the kind of output that might result and how it is to be interpreted, the pitfalls associated with the use of the code, and the uncertainties inherent in the data and the modeling. It is expected, however, that this chapter will also appeal to a wider audience. The layman should be able to find enough qualitative material to give him a good idea of what consequence modeling is about; and the expert, it is hoped, will benefit from the discussion of various topics that are still subject to debate and controversy in the consequence-modeling and scientific communities.

The remainder of this section contains a brief description of the scope of this chapter. The overview in Section 9.2 delineates the major tasks of a consequence analysis, explains why each task is done, what information results, and how it is to be used.

Section 9.3 discusses the various elements of a consequence analysis: (1) transport and diffusion in the atmosphere and/or water; (2) deposition processes; (3) processes that lead to the accumulation of radiation doses; (4) protective measures, such as evacuation, that can reduce radiation doses; (5) the effects of radiation doses on the human body; and (6) economic impacts. Some topics are subject to argument and continuing development since consequence modeling is not a precise science but contains large uncertainties and gaps in knowledge. Where an understanding of the current debate is deemed necessary for a sensible interpretation of the results, a discussion of this debate is included, wind-shift models being a case in point (see Appendix D4). Other areas that are described in some depth are those in which the user's choice of input data can significantly affect the output. A particularly important example is that of evacuation and sheltering (see Appendix E).

Section 9.4 is devoted to the collection and processing of the input data. Section 9.5 is a step-by-step set of procedures for a consequence analysis, starting with the information requirements and ending with the final product, including examples of the results and how they can be displayed. This section also discusses the probabilistic aspects of a consequence analysis in the context of a complete probabilistic risk assessment.

Section 9.6 covers assumptions, sensitivities, and uncertainties. Sections 9.7 and 9.8 describe the methods of documentation and provisions for the assurance of technical quality, respectively. Supporting material is presented in Appendix D, which covers issues in dispersion modeling; Appendix E, which describes evacuation and sheltering; and Appendix F, which discusses liquid pathways.

9.1.2 PURPOSE AND SCOPE OF CONSEQUENCE MODELING

The complete range of calculations carried out in the course of a PRA bridges the gap between the engineering and operations associated with the reactor and the potential risks that the reactor poses to the public. Consequence analysis provides the final link in this chain of calculations and is intended to assess the effect of accidental releases of radionuclides on the surrounding population and the environment.

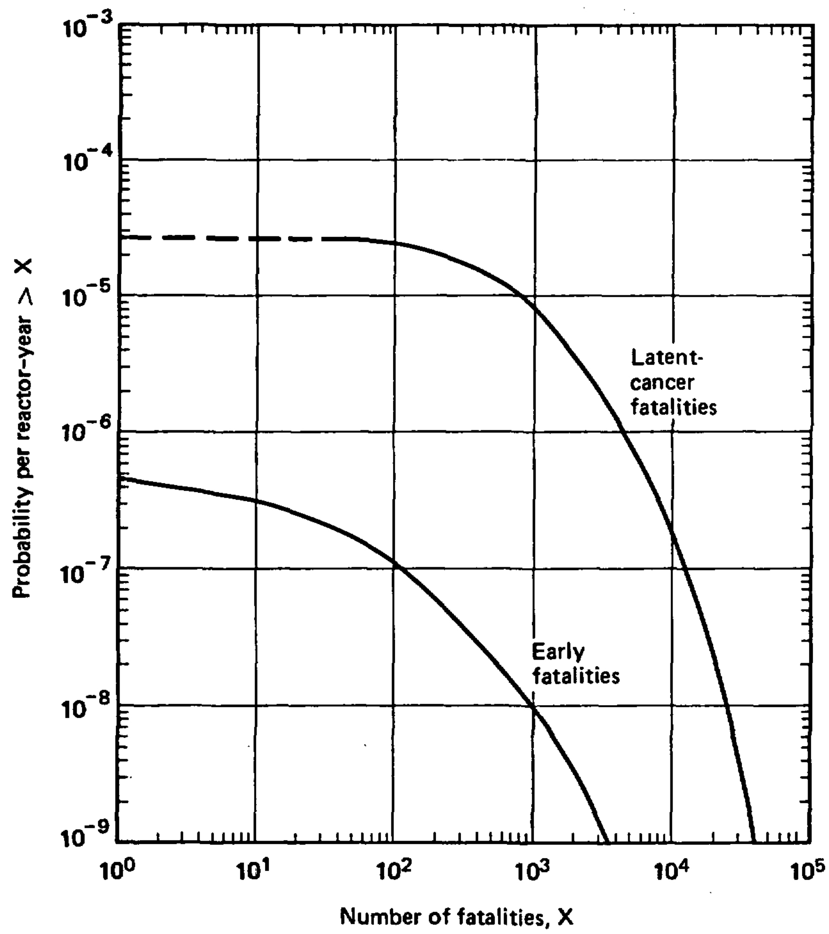
Consequence modeling can therefore be defined as a set of realistic calculations of the ranges (probabilities of occurrence and magnitudes) of adverse impacts that would follow from an accidental release of radionuclides. These adverse impacts, commonly referred to as "public risks," include (1) early and long-term deaths; (2) early and long-term injuries; (3) genetic damage; (4) the contamination of property, land, and water; and (5) economic impacts. These outputs are discussed in more detail in Section 9.5. Consequence modeling provides the means for relating these risks, at both the individual and the societal level, to the characteristics of the radioactive release.

Consequence modeling has many actual or potential applications, including the following examples:

1. Risk evaluation--generic or site specific, societal or individual.
2. Evaluation of alternative design features.
3. Environmental impact assessment.
4. Rulemaking and regulatory procedures.
5. Emergency planning and response.
6. The development of criteria for the acceptability of risk.
7. The provision of focus for research needs.
8. Accident liability.
9. Instrumentation needs and dose assessment.

In the short space allotted to this introduction, it is not possible to describe the application of consequence analysis to each of the topics listed above. The reader may find the examples that follow instructive, however.

<u>Risk Evaluations--Generic or Site Specific</u>. It is usual to present risks as "complementary cumulative distribution functions" (CCDFs), and two examples are given in Figure 9-1, which shows the predicted probability per reactor-year (frequency) with which an accident might occur and cause the deaths of as many as, or more than, the corresponding number of people. These CCDFs, which are probably the most natural form of the output of a



Note: Approximate uncertainties are estimated to be represented by factors of 1/4 and 4 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

Figure 9-1. Frequency distribution for early fatalities and latent-cancer fatalities. From the Reactor Safety Study (USNRC, 1975).

consequence analysis, are taken from the most celebrated of all risk assessments performed to date, the Reactor Safety Study (RSS--USNRC, 1975). The CCDFs themselves can be used as a measure of public or societal risk. Some authors take the integrals under the CCDFs, which are generally approximately equal to the expected (in the statistical sense) number of early or latent fatalities per year, and use these figures as a measure of public risk. The CCDFs and/or the expected values can then be compared with similar quantities for other industrial activities, in order to put them in perspective.

The Reactor Safety Study was a generic study. Numerous site-specific studies are also under way or have recently been completed, such as that for the Limerick site (Philadelphia Electric Company, 1981), the Zion PRA (Commonwealth Edison Company, 1981), and the German Risk Study (Gesellschaft fuer Reaktorsicherheit, 1980; EPRI, 1981).

Rulemaking and Regulatory Procedures. A good example of this use of consequence modeling is the current Sandia Nuclear Power Plant Siting Study (Strip et al., 1981), which is intended to assist the U.S. Nuclear Regulatory Commission (NRC). The NRC is currently revising its regulations on the siting of nuclear power plants and has asked Sandia National Laboratories to provide technical guidance for establishing (1) numerical criteria for the population density and distribution around the sites of nuclear power plants and (2) standoff distances for offsite hazards. In order to provide this guidance, calculations have been performed to address the following questions:

1. What range of risk is associated with currently existing sites?

2. What characteristics of the surrounding population (distance, distribution) influence risk?

3. What impacts do other site and design characteristics have on risk?

4. What is the influence of emergency response on risk?

These calculations included population data for 91 sites at which reactors are operating or under construction and sensitivity analyses to examine the influence on risk of emergency-response alternatives, weather conditions at the site, release characteristics (including release fractions), reactor power level, and reactor design.

A report describing the Sandia Nuclear Power Plant Siting Study in detail should be available soon. It is a good example of a study of its kind because it combines several of the applications of consequence analysis and also contains a sensitivity study.

Further examples of the use of consequence analyses in the regulatory context can be found in the recent series of supplements to the environmental reports for several reactor sites (see, for example, USNRC, 1981). These supplements have been produced by the NRC in order to fulfill its interpretation of the requirements of the National Environmental Policy Act, as put forward in the Commission's Statement of Interim Policy:

> ...Environmental Impact Statements shall include consideration of the site-specific environmental impacts attributable to accident sequences that lead to releases of radiation and/or radioactive materials, including sequences that can result in inadequate cooling of the reactor fuel and to melting of the reactor core. In this regard, attention shall be given both to the probability of occurrence of such releases and to the environmental consequences of such releases.

To implement this policy, the NRC has carried out site-specific probabilistic consequence analyses of Class 9 accidents.

The Sandia Nuclear Power Plant Siting Study and the NRC's applications of consequence modeling in environmental impact statements are examples of uses of consequence analyses that do not fit into the PRA categories defined in Chapter 2. In each of these studies, a generic source term* was used. Thus, there are uses in which a consequence analysis can be, and has been, carried out outside the context of a PRA.

Emergency Planning and Response. Several recent studies have used the RSS consequence model, CRAC, for guidance in emergency planning and response. One such study (Aldrich, McGrath, and Rasmussen, 1978; Aldrich et al., 1978) examined the relative merits of evacuation and sheltering followed by population relocation as protective measures for core-melt accidents, the distances to which (or areas within which) they might be needed, and the time available for their implementation. Partly on the basis of this analysis, the NRC has required the implementation of emergency-planning zones for the plume-exposure pathway, with a radius of approximately 10 miles, for all operating plants in the United States (Collins et al., 1978).

Another study has been performed to provide guidance to policy makers concerning (1) the effectiveness of potassium iodide as a blocking agent in potential reactor-accident situations, (2) the distance to which (or the area within which) it should be distributed, and (3) its relative effectiveness in comparison with other available protective measures (Aldrich and Blond, 1980, 1981). Again, the analysis was performed with the RSS consequence model. The conclusion was that potassium iodide does not appear to be a cost-effective protective measure.

Evaluation of Alternative Design Features. Carlson and Hickman (1978) considered a number of design alternatives for light-water reactors (LWRs): (1) stronger containment, (2) shallow underground siting, (3) deep underground siting, (4) increased containment volume, (5) filtered atmospheric venting, (6) compartment venting, (7) thinned basemat, (8) evacuated containment, and (9) double containment. For each of these alternatives, they carried out a consequence analysis and calculated the integrals under the CCDFs for early fatalities, latent-cancer fatalities, and property damage. These results were then used as a basis for estimating the cost effectiveness of each design alternative.

The list of the uses of consequence analysis given above, together with the examples that follow it, should give the reader a good idea of the range of applications of a consequence analysis.

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*In the Sandia study, there are five source terms ranging from a gap-activity release to a core melt with a large radionuclide release directly to the atmosphere (Aldrich et al., 1981a). These source terms were developed by the NRC specifically for the siting study. In the consequence analyses for the environmental impact statements, the NRC used the "rebaselined" pressurized-water reactor (PWR) or boiling-water reactor (BWR). These are essentially representations of the Surry PWR or the Peach Bottom BWR, which were the reactors analyzed in the Reactor Safety Study, with some modifications to account for calculations and reanalyses carried out since the report of the Study was written. Each environmental impact statement contains an appendix describing the appropriate rebaselined reactor.

9.2 OVERVIEW

There are a number of tasks involved in a consequence analysis, as outlined below.

1. Acquiring background. A beginner must first acquaint himself with what is typically done in a consequence analysis and with the various codes that are available. Once this has been done, he can make an intelligent choice of code for his own use.

2. Deciding on the purpose of the analysis. This decision is important. It influences the choice of code, the requirements for input data, and the choice of output.

3. Choosing a computer code for consequence modeling.

4. Computer-code debugging and/or modification. The purpose of the calculations may require modifications to the code.

5. Collecting input data. For the user of consequence-modeling codes, this is the most important task in the analysis. It offers him the chance to make a significant impact on the results of the calculations.

6. Exercising the code. In principle, this is straightforward, assuming that the ground has been well prepared by the conscientious performance of the earlier tasks. This task also includes any sensitivity studies that may be carried out as part of an uncertainty analysis.

7. Report writing and interpretation of results.

The experience of the consequence modeler will, of course, determine which of these tasks he needs to do. A complete beginner would start with task 1. An experienced member of a PRA team would need to carry out task 2, deciding on the purpose of a consequence analysis, but could then begin with task 5, the collection of input data.

9.2.1 TASK 1: BACKGROUND STUDY

As an introduction to the subject, Appendix VI of the Reactor Safety Study contains a comprehensive survey of all of the essential elements. In order to understand the meteorological modeling, Meteorology and Atomic Energy--1968 (Slade, 1968) is a thorough review that will be shortly updated and retitled Atmospheric Science and Power Production. As examples of the use of existing codes in recent risk assessments, it is instructive to review the Limerick study (Philadelphia Electric Company, 1981), the German Risk Study (Gesellschaft fuer Reaktorsicherheit, 1980; an English translation has been prepared by EPRI, 1981), and the Zion study (Commonwealth Edison Company, 1981).

Figure 9-2 gives a schematic outline of the arrangement of the computational elements or submodels of a typical consequence-modeling code. Most codes are made up of similar elements. Each of the submodels is discussed briefly below, drawing heavily on material contained in the Overview of the Reactor Safety Study Consequence Model (Wall et al., 1977).

9.2.1.1 Description of Radionuclide Release

The calculation begins with a description of the characteristics of the radionuclide release, including the quantity of each radionuclide released to the environment, the amount of energy associated with the release, the duration of the release, the time of the release after accident initiation, the warning time for evacuation, and the frequency of occurrence predicted for the accident. An example of this kind of input data, generated by the engineering analysis of the PWR and BWR reactors examined in the Reactor Safety Study, appears in Table 9-1. This input is discussed more fully in Section 9.4.2.

9.2.1.2 Atmospheric Dispersion and Weather Data

Most consequence-modeling codes simulate the atmospheric dispersion of the released radioactive material by using a Gaussian dispersion model to calculate ground-level instantaneous and time-integrated airborne concentrations and deposited levels of radioactivity. This is done as a function of time and of distance from the reactor. In the consequence-modeling codes available in the United States, the Gaussian model is generally used in such

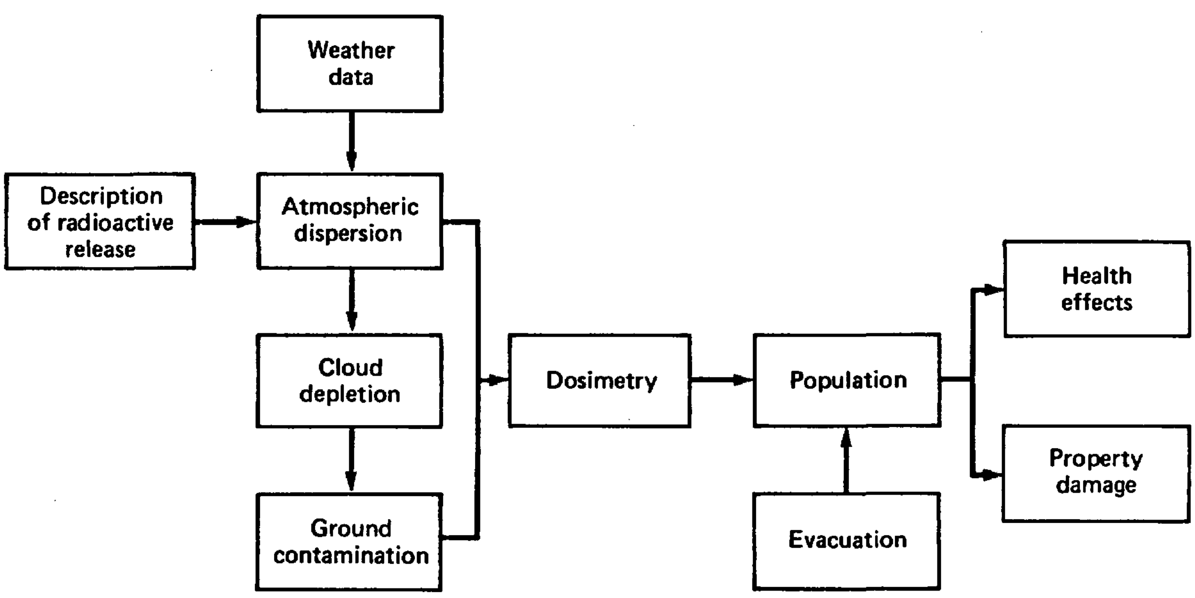

Figure 9-2. Schematic outline of a typical consequence model. From Wall et al. (1977).

Table 9-1. Summary of RSS release categories for hypothetical accidents[a]

Release category[d]	Probability per reactor-yr	Time of release (hr)	Duration of release (hr)	Warning time for evacuation (hr)	Elevation of release[b] (meters)	Energy of release (10^6 Btu/hr)	Fraction of core inventory released[c]: Xe-Kr	I[e]	Cs-Rb	Te-Sb	Ba-Sr	Ru[f]	La[g]
PWR-1	9×10^{-7}	2.5	0.5	1.0	25	20 and 520[h]	0.9	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR-2	8×10^{-6}	2.5	0.5	1.0	0	170	0.9	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR-3	4×10^{-6}	5.0	1.5	2.0	0	6	0.8	0.2	0.2	0.3	0.02	0.03	3×10^{-3}
PWR-4	5×10^{-7}	2.0	3.0	2.0	0	1	0.6	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-4}
PWR-5	7×10^{-7}	2.0	4.0	1.0	0	0.3	0.3	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}	7×10^{-5}
PWR-6	6×10^{-6}	12.0	10.0	1.0	0	NA	0.3	8×10^{-4}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR-7	4×10^{-5}	10.0	10.0	1.0	0	NA	6×10^{-3}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
PWR-8	4×10^{-5}	0.5	0.5	NA[i]	0	NA	2×10^{-3}	1×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0	0
PWR-9	4×10^{-4}	0.5	0.5	NA	0	NA	3×10^{-6}	1×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0	0
BWR-1	1×10^{-6}	2.0	0.5	1.5	25	130	1.0	0.40	0.40	0.70	0.5	0.5	5×10^{-3}
BWR-2	6×10^{-6}	30.0	3.0	2.0	0	30	1.0	0.90	0.50	0.30	0.10	0.03	4×10^{-3}
BWR-3	2×10^{-5}	30.0	3.0	2.0	25	20	1.0	0.10	0.10	0.03	0.01	0.02	4×10^{-3}
BWR-4	2×10^{-6}	5.0	2.0	2.0	25	NA	0.6	8×10^{-4}	5×10^{-3}	4×10^{-3}	6×10^{-4}	6×10^{-4}	1 x 10-4
BWR-5	1×10^{-4}	3.5	5.0	NA	150	NA	5×10^{-4}	6×10^{-11}	4×10^{-9}	8×10^{-12}	8×10^{-14}	0	0

[a]From Wall et al. (1977).

[b]A 10-m elevation is used in place of zero representing the midpoint of a potential containment break. Any impact on the results would be slight and conservative.

[c]Background on the isotope groups and release mechanisms is presented in the Reactor Safety Study, Appendix VII (USNRC, 1975).

[d]The definition of release categories is discussed in Section 9.4.2.9.

[e]Organic iodine is combined with elemental iodine in the consequence calculations. Any error is negligible since the release fraction of organic iodine is relatively small for all large release categories.

[f]Includes Ru, Rh, Co, Mo, Tc.

[g]Includes Y, La, Zr, Nb, Ce, Pr, Nd, Np, Pu, Am, Cm.

[h]Accident sequences within the PWR-1 category have two distinct energy releases that affect consequences. The PWR-1 category is subdivided into PWR-1A, with a probability of 4×10^{-7} per reactor-year and an energy of release of 20×10^6 Btu/hr; and PWR-1B, with a probability of 5×10^{-7} per reactor-year and an energy of release of 520×10^6 Btu/hr.

[i]Not applicable.

a way as to allow changes in atmospheric stability, wind speed, and precipitation for each successive hour of travel time. Some codes also allow the wind direction to change. The hourly weather data required as input are usually generated by processing data collected at the reactor site itself or at nearby weather stations.

In general, consequence-modeling codes simulate the behavior of the radioactive plume as it travels tens or hundreds of kilometers downwind--that is, for many hours during which the weather conditions may change. In principle, there may be a different sequence of hourly weather changes for each of the 8760 hours during a full year at which the accident might take place. In practice, it is usually prohibitively expensive to run each of these sequences in turn, and some method must be devised for selecting a sample. In some codes this can be done randomly, in others by selecting starting times that are equally spaced throughout the year. Another possibility is to first combine the weather sequences into groups in which the pattern of hourly weather changes is similar and then to ensure that the sampling process covers all of the groups. This question of how best to sample weather data is important and is discussed more fully in Appendix D4.1.2. Some other important issues, such as the differences between codes that do or do not allow changes in wind direction as the plume travels downwind, are also addressed in Appendix D4.

The basic Gaussian model is modified to take into account a number of phenomena. Among them are radioactive decay and daughter buildup, which are treated in ways that can be found in any standard textbook. Allowance is usually made for the mixing of the radioactive plume as it emerges into the turbulent wake of the reactor building. The atmospheric boundary layer, which is the layer of turbulent air adjacent to the surface of the earth, is almost always capped by an overhead inversion, which is a layer of very stable air that acts as an effective barrier to the upward dispersion of the plume. The height of the base of this layer, often termed "the inversion lid," depends on several phenomena, including the intensity of turbulence in the layer of air beneath it, which in turn depends on the time of day and the wind speed. Methods of treating the inversion lid as a function of time can become quite sophisticated. (See Appendix D4 for a further discussion.)

If the plume is buoyant, it is allowed to rise according to standard procedures available in the literature. Some codes allow the plume to penetrate the inversion lid.

9.2.1.3 Deposition--Ground Contamination

As the plume of radioactive material travels outward from the reactor, various mechanisms remove the airborne material. In addition to radioactive decay, the radioactive material is removed by such deposition processes as impaction on obstacles (dry deposition) and by precipitation scavenging (wet deposition).

These deposition mechanisms cannot be specified precisely. There are significant dependences of removal rates on, among other things, the type and rate of precipitation, particle density and size distribution, the surface characteristics of the ground, and weather conditions. For simplicity, the dry-deposition velocity (ratio of the deposition flux to the air concentration at a particular distance from the surface) is assumed to be constant for particulate matter. When it rains or snows, wet deposition occurs simultaneously with dry deposition. Wet deposition is modeled by a simple exponential removal rate, which should be dependent on the rate of rainfall. When the occurrence of precipitation is specified by the weather data, it is assumed to occur uniformly within time and throughout the spatial interval in which the plume is located. The removal rate is a function of the thermal stability. The noble gases are assumed to be insoluble and nonreactive, and therefore are not removed by either dry or wet deposition.

The ground concentration is calculated from the air concentration and the deposition rate. The material deposited on the ground is subtracted from the airborne material.

Both dry deposition and wet deposition are still matters of considerable discussion among consequence modelers. Dry deposition is discussed more fully in Appendix D3.

9.2.1.4 Processes That Lead to the Accumulation of Radiation Doses (Dosimetry)

Using the procedures described above, for each selected accident starting time, spatial distributions of instantaneous and time-integrated airborne concentrations and deposited levels of radioactive material are estimated. These quantities are then used to calculate the potential radiation doses that would be received by individuals and populations--doses that could be accumulated in a number of ways. Figure 9-3 shows some of these possible pathways by which radioactivity could reach people. (This figure is not intended to be a comprehensive summary.) It is convenient to classify the exposure pathways as those associated with the passing cloud and those associated with ground contamination.

The airborne radioactive material leads to radiation doses caused by external radiation from the plume ("cloudshine") and radiation from inhaled radionuclides. To receive the external radiation, a person must be either immersed in the plume or in its general vicinity. The consequence-modeling code relates the concentration of radioactive material in the air to an external dose delivered to various body organs (e.g., bone marrow, gastrointestinal tract).

The radiation dose from inhaled radioactive material is proportional to the exposure to the airborne concentration of radionuclides at roughly 2 meters above the ground and to the individual's breathing rate. The dosimetric model used to derive the dose-conversion factors describes the time-dependent movement of the radioactive material within the body. An important element of the model from which the data used in most consequence-analysis codes are derived is the well-known lung model of the International Commission for Radiological Protection (ICRP, 1966).

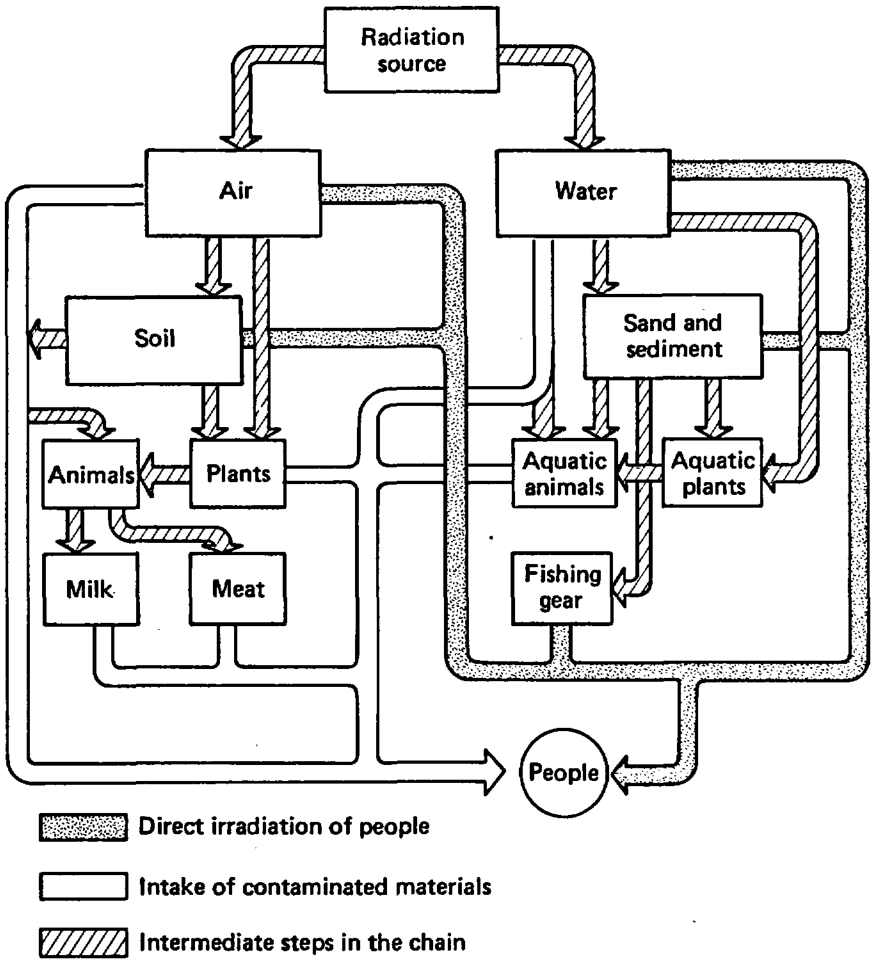

Figure 9-3. Examples of radiation pathways. From Safety and Nuclear Power, United Kingdom Atomic Energy Authority, London, England.

In essence, the radioactive material deposited on the ground delivers radiation doses through three pathways: external irradiation due to gamma rays emitted by deposited material ("groundshine"), the inhalation of resuspended radioactive material, and the ingestion of contaminated food and water. The ingestion of the radioactive material may result from direct deposition onto vegetation, which is consumed by people or by animals furnishing food for people, or from the more indirect pathways involving the uptake of ground-deposited radioactive material through the roots of plants.

9.2.1.5 Population Distribution

Once the radiation doses delivered to individuals have been calculated, they must be combined with the population distribution. In general, consequence models assign the population to a grid consisting, first, of a number of sectors. Within each sector, radial intervals are defined. The population within a sector and between two radial intervals is effectively assumed to be uniformly distributed. Population data are usually obtained by processing U.S. Census data, carrying out house counts, and examining aerial photographs. Some users extrapolate the census data to plant midlife.

9.2.1.6 Evacuation and Other Measures That Reduce Radiation Doses

Evacuation is the expeditious movement of people to avoid or reduce immediate exposure to the passing cloud. It is in the choice of such parameters as the delay time (the period between the declaration of a general emergency by the plant emergency director and the time at which evacuation actually begins) and the effective evacuation speed that the user can profoundly influence the results of his calculations. This is particularly true of the predicted numbers of early fatalities and early injuries, which are very sensitive to the radiation dose accumulated through exposure to gamma rays emitted by deposited fission products during the first few hours after the accidental release of radioactivity has taken place.

It is very important that the evacuation model be sensibly handled by the user of the code. For this reason, Appendix E presents a thorough description of some evacuation models and an in-depth discussion of the input-data requirements.

It is assumed in consequence modeling that people will take advantage of structures in the neighborhood of reactors in order to shelter from external irradiation by gamma rays. Gamma rays emitted by the passing cloud (cloudshine) are attenuated by, for example, the walls of buildings; gamma rays emitted by deposited radioactive material (groundshine) are attenuated both by buildings and by surface rugosities. Consequence-modeling codes require shielding factors for both cloudshine and groundshine for people assumed to be using shelters. Also required may be shielding factors for people waiting to evacuate, people evacuating, and people behaving normally. Appendix E explains how to calculate such shielding factors.

Another measure that can be used to reduce radiation doses is relocation. This is the permanent or long-term removal of people from a contaminated area in order to reduce the radiation dose accumulated by long-term exposure to the deposited radioactive material.

The countermeasures treated in many consequence-modeling codes also include interdiction and decontamination. The radioactive contamination of a large area may result in the contamination of milk produced by cattle grazing on contaminated pastures, in the external contamination of crops, and/or in excessive radiation doses to people. In such events, the milk and crops may be impounded and/or the people relocated for a period of time. All of these actions are called "interdiction."

The interdiction model is based on the concept of maximum acceptable doses. The dose criteria used in the Reactor Safety Study (USNRC, 1975) were based on the recommendations of the U.S. Federal Radiation Council (FRC, 1965) and the British Medical Research Council (MRC, 1975).

The dose criteria are translated into corresponding contamination levels (curies per square meter) of different radionuclides by dosimetric models like those described in the Reactor Safety Study and an environmental model that incorporates the grass-cow-man or soil-root-crop-man pathways. Since the milk interdiction level is the most restrictive, the area over which milk would be impounded would be the largest. Conversely, the interdiction level for human occupancy is the least restrictive, and therefore the area from which people would be relocated would be the smallest. The "weathering" of deposited radionuclides is also incorporated so that the interdiction distance slowly moves toward the reactor.

Decontamination is defined as the cleanup and removal of radionuclides (see Section 9.3.4.4). A measure of the effectiveness of decontamination operations is the decontamination factor--that is, the original concentration of the contaminant (in curies per square meter) divided by its concentration after decontamination. The decontamination model is illustrated in Figure 9-4. Without decontamination, the interdiction criterion translates to a distance R_1. With a maximum decontamination factor of 20, the land area between R_1 and R_2 will become available for reoccupation. Subsequent weathering of the radionuclides will reopen the land area between R_2 and R_3.

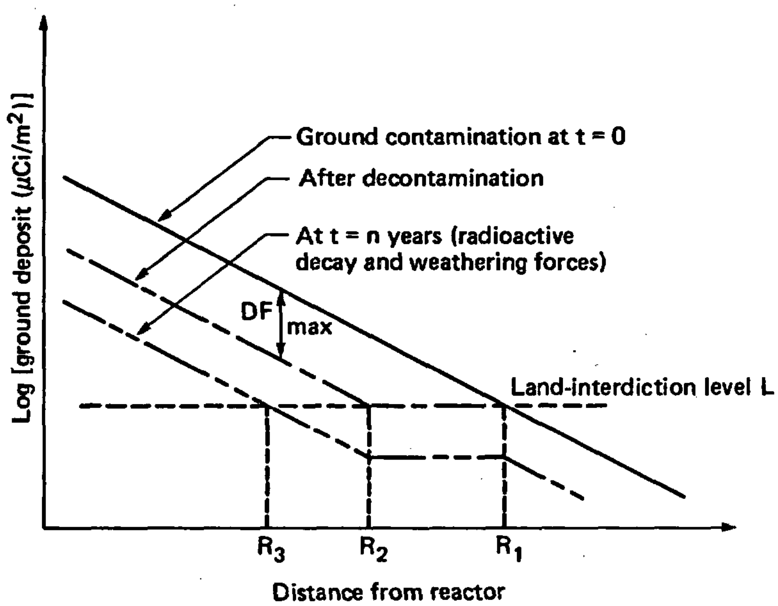

Figure 9-4. Illustrative decontamination model for ground-level releases. From Wall et al. (1977).

9.2.1.7 The Effect of Radiation on the Human Body (Health Effects)

Three categories of potential health effects may be calculated: early and continuing somatic effects, late somatic effects (cancers), and genetic effects. Early and continuing somatic effects manifest themselves within days up to a year after exposure. By contrast, latent cancers would probably be observed from at least 2 to 40 years after exposure and genetic effects in succeeding generations.

Early and continuing fatalities may result from radiation damage to the bone marrow, the lung, or the gastrointestinal tract. Past studies indicate that radiation damage to the bone marrow is the most important contributor, λ, given the inventory of radionuclides likely to be released to the atmosphere in the event of an accident in a light-water reactor. The relationships between the radiation dose to these various organs and the probability of death (dose-risk or dose-response relationships) are discussed in Section 9.3.5 and reviewed in depth in Appendix VI of the Reactor Safety Study. The RSS also estimated the number of prenatal deaths and of early injuries, including hypothyroidism, temporary sterility, congenital malformations, growth retardations, cataracts, and prodromal vomiting. In general, it is not necessary to consider early effects in such detail. The predicted numbers of early fatalities and injuries are usually sufficient to give an adequate notion of the public risks associated with early effects.

Late somatic effects consist of latent-cancer fatalities, nonfatal cancers, and benign and cancerous thyroid nodules. After the irradiation of a large number of people, there is generally a latent period during which no increase in cancer incidence is detectable. After this period, the radiation-induced cancers tend to appear at an approximately uniform rate for a period of years, which is termed the "plateau." The plateau period could in some cases extend over the lifetime of the individual. The dose-response relationships for cancer induction are discussed in Section 9.3.5 and in Appendix VI of the Reactor Safety Study.

9.2.1.8 Economic Costs (Property Damage)

Property damage after a postulated reactor accident is not of the same nature as that resulting from most other potential catastrophic events (i.e., there is no physical damage to offsite property). The damage arises from contamination with radioactive material and the possible radiation dose that could be received if the property were used in its intended manner. The restriction in the use of the property results in economic loss.

The components of property damage, as assumed and modeled in the consequence model, are evacuation costs, loss of agricultural products, decontamination costs, and population-relocation costs. The main problem is to ensure that realistic figures are used for the various elements of these costs (see Section 9.4).

9.2.2 TASK 2: DECIDING ON THE PURPOSE OF THE CONSEQUENCE CALCULATIONS

Before embarking on the choice of a consequence-modeling code and its use, it is necessary to take some time to think of the output that is required and the purpose for which it is to be used. This influences, for example, which code is to be used and what input data are required. Some examples are as follows:

1. In the recent Zion study (Commonwealth Edison Company, 1981), it was deemed sufficient to calculate CCDFs for early fatalities and

injuries, latent-cancer fatalities, and the population dose (man-rem). This removed the requirement for the collection of data pertaining to economic costs.

2. A study like Sandia's Nuclear Power Plant Siting Study (Strip et al., 1981) would require the handling of large amounts of meteorological and population data from a considerable number of sites.

3. The NRC has begun to publish supplements to the environmental impact statements for various reactors in response to the requirements of the National Environmental Policy Act (NEPA). These supplements contain fairly stylized calculations. Any utility that wishes to respond to the requirements of NEPA would presumably deem it sufficient to do similar calculations.

4. Organizations wishing to carry out a detailed analysis of evacuation procedures, taking into account existing road networks, might consider looking at a code that is capable of mapping the road network.

The above examples demonstrate the importance of having a clear idea of the purpose and the required output of a consequence analysis at a very early stage.

9.2.3 TASK 3: CHOICE OF CODE FOR CONSEQUENCE MODELING

The potential user of consequence models may wish to be told categorically that code X is manifestly the best that is available and should be used in preference to all others. Unfortunately, consequence modeling is as much an art as a science. There are large gaps in knowledge that can be filled only by the judgment of the modeler. The area is still being developed, with new and promising changes to codes. Often it is how intelligently the code is used, rather than which particular one, that determines whether the results are meaningful or not.

In the United States, there are four codes that can be used for a complete consequence analysis. The reader may be inclined to object that there are many more than four such codes. To be precise, there are four codes that both contain all of the necessary elements of a consequence model and perform the probabilistic manipulations that are necessary for the calculation of CCDFs. Other codes may contain many excellent and sophisticated features, but they are not fully developed consequence-modeling codes. The four in question are CRAC, the code used during the RSS, and three offshoots, CRAC2, CRACIT, and NUCRAC.

<u>CRAC</u>. The code CRAC (Calculation of Reactor Accident Consequences) was developed for the Reactor Safety Study (USNRC, 1975, Appendix VI). It was the first code to integrate all of the elements of a consequence model into a package capable of generating CCDFs and contains what were, at the time, innovative features (in the context of consequence modeling), such as the treatment of changing weather conditions and the incorporation of chronic pathways.

CRAC2. The CRAC2 code is a revision of CRAC (Ritchie et al., 1981a). Recently issued by Sandia National Laboratories, it incorporates significant improvements in the area of weather-sequence sampling (Ritchie et al., 1981b) and emergency response (Aldrich et al., 1978; Aldrich, Ritchie, and Sprung, 1979).

Weather data are normally collected at reactor sites at hourly intervals. Weather sequence sampling is the selection of a limited number of starting times for accident sequences from the 8760 that are possible in a full year, in order to reduce computing time. CRAC employs a stratified sampling technique whereby weather sequences are selected every 4 days plus 13 hours to cover diurnal, seasonal, and 4-day weather cycles. In this manner, 91 sequences are chosen to represent a year of data. Sensitivity studies performed with CRAC indicate considerable uncertainty in the predicted results, attributable to sampling by this method.

CRAC2 uses a new weather-sequence sampling method that greatly reduces the uncertainty attributable to sampling. Before sampling sequences, the entire year of data is sorted into 29 weather categories, or bins. Categories include sequences in which either rainfall or wind-speed slowdowns occur within specified distance intervals from the plant. Atmospheric-stability and wind-speed categories are also considered. The probability of each weather category is estimated from the number of sequences in the category. Sequences are then sampled from each of the 29 categories (and weighted with appropriate probabilities) for use in risk calculations, thus ensuring that low-probability adverse weather conditions (e.g., rainfall, wind-speed slowdowns) are adequately included.

The emergency-response model in CRAC2 is considerably more realistic than that in CRAC. In CRAC, evacuation was assumed to commence immediately upon warning and to proceed at a very slow speed. Any person overtaken by the plume was assumed to be exposed to the full extent of the plume and to receive a 4-hour ground dose. In contrast, the CRAC2 model includes a delay time between warning and the start of evacuation, more reasonable evacuation speeds, and an explicit calculation of the time during which people are exposed to airborne and deposited radionuclides. The revised model also allows the user to consider population sheltering.

A number of refinements in the calculation of plume rise, washout, and atmospheric dispersion were also incorporated into CRAC2.

CRACIT. Developed by Pickard, Lowe and Garrick, Inc., CRACIT (CRAC Including Trajectories) incorporates major modifications in the atmospheric-dispersion and evacuation models that permit some unique features of a site to be considered (Woodard and Potter, 1979; Commonwealth Edison Company, 1981). The atmospheric-dispersion model in CRACIT used the "modified potential flow" (MPF) method developed by Lantz and Coats (1971). The MPF method incorporates the effect of site-specific topographic features by using digitized terrain data to calculate a temporally and spatially dependent wind field. Using the calculated wind field, CRACIT solves the set of transport and diffusion equations by numerical methods and is thus more realistic than the Gaussian plume model. In CRACIT, the numerical solution is used only to a maximum distance of 14.5 km; the model used beyond this distance is a segmented Gaussian-plume model that incorporates changes in wind direction by changing the trajectory of the plume.

The evacuation model in CRACIT takes into consideration the likely evacuation routes at a site (the CRAC and CRAC2 models assume evacuees move radially away from the reactor) as well as traffic jams that may occur during an evacuation. CRACIT also contains a number of additional refinements in the calculation of atmospheric dispersion, plume rise, washout, and interactions between the plume and the inversion layer.

Because it calculates a three-dimensional wind field, performs numerical dispersion calculations, and incorporates an actual road network into the evacuation model, CRACIT requires considerably more input data and computation time than does CRAC.

NUCRAC. NUCRAC, developed by Science Applications, Inc., incorporates major modifications in two areas: plume depletion by dry deposition and chronic-exposure pathways (Kaul et al., 1980; Kaul, 1981a). The model allows for a distribution of particle sizes in the material released from the containment. Dry deposition is modeled with Overcamp's (1976) surface-depletion method, which takes into account the gravitational settling of particles on the basis of particle size. NUCRAC, however, does not currently consider plume depletion by wet deposition. The improved model of chronic-exposure pathways in NUCRAC treats a larger number of radionuclides and better reflects the site-specific details of agricultural production.

Other Codes. A document that will discuss the full range of consequence models available worldwide and their capabilities is the forthcoming report of the International Benchmark Comparison of Reactor Accident Consequence Models,* henceforth referred to as the "Benchmark." This document, when it becomes available, should be made required reading for all would-be users of consequence models, particularly those who wish to interpret the results generated by consequence-modeling codes and to assess the impact of uncertainties. Examples of other consequence-modeling codes are TIRION, developed by the United Kingdom Atomic Energy Authority (Kaiser, 1976; Fryer and Kaiser, 1979), ALICE, developed by the French Commissariat à l'Energie Atomique (Maire et al., 1981) and the Finnish code ARANO (Nordlund et al., 1979).

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*The Benchmark exercise is being carried out under the aegis of the Committee for the Safety of Nuclear Installations (CSNI), Nuclear Energy Agency, Organization for Economic Cooperation and Development. The exercise has consisted of the definition of a number of standard problems and their solution by some 20 participants from Europe, Japan, and the United States, using their own consequence models. The results are to be presented in a forthcoming report together with interpretation by various problem coordinators. Preliminary presentations on the activities of the Benchmark are given by Blond et al. (1981) and Aldrich et al. (1981b). Detailed specifications of the Benchmark problems, including site and release characteristics, are available on request from Division 9415, Sandia National Laboratories, Albuquerque, N.M. 87185.

The user should be aware that the state of the art of consequence modeling is one of change. He should always be on the lookout for improvements because models are continually being updated.

9.2.4 TASK 4: CODE DEBUGGING AND MODIFICATION

Once the code has been obtained, the tedious but necessary process of debugging it and making sure that it can run on the user's machine must be undertaken. At the same time, any necessary modifications to the code should be carried out--modifications designed to produce additional output data, for example.

9.2.5 TASK 5: COLLECTION OF INPUT DATA

Consequence-modeling codes are elaborate and require what seem to be endless amounts of input data. This is where the user can have a considerable impact on the results, and his choice of certain inputs will determine whether the results are meaningful or not. An example that has already been mentioned is the choice of delay time for evacuation.

In general, it is the user's responsibility to collect and process data in some or all of the following areas: (1) input from the calculations of radionuclide release and transport (e.g., magnitude, duration and rate of release, energy of release, frequencies); (2) population and meteorological data; (3) economic data; (4) health-physics data; (5) emergency-response information; and (6) criteria for interdiction and decontamination. Section 9.4 describes where to obtain such data and how to process it. It is to be emphasized that data collection is a time-consuming procedure and must be started at an early stage in a consequence analysis, considerably in advance of running the code.

9.2.6 TASK 6: EXERCISING THE CODE

This is usually the most straightforward part of the consequence-analysis calculation. It is necessary to carry out runs of the code for the various cases needed to generate the required CCDFs or other results and to repeat some of the runs for changes in some of the parameters (e.g., evacuation speed, deposition velocity) for which sensitivity studies are thought to be desirable. These sensitivity studies may be used as the basis for an uncertainty analysis.

9.2.7 TASK 7: REPORT WRITING AND INTERPRETATION OF RESULTS

Once the results have been completed, it is necessary to describe what has been done, perhaps in the form of a report like that outlined in Section 9.7. Included in the report will be the display and interpretation of the results, as described in Section 9.6.

9.3 METHODS

This section describes some of the more common methods used in consequence modeling.

9.3.1 RADIONUCLIDE TRANSPORT AND DIFFUSION

According to Figure 9-2, the first step in the chain of calculations is the release of radioactivity into the atmosphere or into water. The atmospheric pathway is generally the most important in the case of nuclear reactors, but there are postulated circumstances--for example, core melt-through into an aquifer, or the release of sumpwater--in which the water pathway should be considered. In general, the water pathway is not treated on the same footing as the atmospheric pathway in consequence modeling, because it can be shown to be a less significant contributor to the magnitude of the predicted consequences. Liquid-pathway modeling has been examined in two recent reports (USNRC, 1978; Niemczyk et al., 1981). The state of the art of water-pathway modeling is summarized in Appendix F, but the subject is not treated further in this section.

9.3.1.1 <u>The Gaussian Plume Model</u>

The most commonly used model of atmospheric dispersion in consequence-modeling codes is the Gaussian one. Appendix D shows that this popularity arises for the following reasons:

1. Economical use of computer time.

2. General lack of availability of the meteorological parameters necessary for input to more complicated models.

3. Evidence that in some circumstances, such as dispersion over flat terrain, the results do not differ sufficiently from those of more-complicated models to make the use of the latter worthwhile in analyses that require repeated use of the meteorological model.

The conventional Gaussian formula for the time-integrated concentration χ at the point (x,y,z) is

$$\chi(x,y,z) = \frac{Q \exp\left[-y^2/2\sigma_y^2(x)\right]}{2\pi\sigma_z(x)\ \sigma_y(x)\bar{u}} \left\{ \exp\left[-\frac{(z+h)^2}{2\sigma_z^2(x)}\right] + \exp\left[-\frac{(z-h)^2}{2\sigma_z^2(x)}\right] \right\} \quad \text{Ci-sec/m}^3 \qquad (9\text{-}1)$$

where the pair of exponentials summed inside the braces expresses the fact that total reflection at the ground has been assumed. The symbols are defined as follows:

Q = the total amount of effluent emitted (curies).

h = the height of the source (meters).

$\sigma_z(x)$, $\sigma_y(x)$ = the vertical and horizontal standard deviations (meters), respectively.

x = the distance downwind (meters).

y = the distance across wind (meters).

z = the height above the ground (meters).

$\bar{u}$ = the mean wind speed (m/sec).

Since the wind speed varies with height, it is not possible to define $\bar{u}$ unambiguously. In many experiments, $\bar{u}$ is the wind speed at the height of the source or at the height of a nearby tower. Smith and Singer (1965) show that a reasonable estimate of $\bar{u}$ is obtained by calculating the wind speed at a height $0.62\sigma_z(x)$. This conclusion, however, is model dependent. Unless otherwise stated, it is assumed in this chapter that $\bar{u}$ is the wind speed measured at a height of 10 meters, that is, $\bar{u} = \bar{u}(z = 10) = \bar{u}(10)$. Clarke et al. (1979) say that the product $\sigma_y\bar{u}$ tends to remain constant with in creasing height, so that it is acceptable to use $\bar{u}(10)$, provided that appropriately measured σ_y values are also taken. Some computer codes (e.g., CRACIT) allow $\bar{u}$ to change with height.

The use of the Gaussian model can be justified in a qualitative way by appealing to the random properties of atmospheric turbulence. A small particle of radioactive material, while being carried downwind at the mean wind speed, is also thrown about at random by the turbulent forces acting upon it; that is, it can be regarded as taking a random walk. As is well known, the distribution of a large number of such particles, each of which has taken a large number of random steps, can be described by the Gaussian formula.

Some authors approximate the lateral spreading of diffusing plumes by a "top-hat" distribution. In the Reactor Safety Study (USNRC, 1975, Appendix VI), the quantity

$$\frac{1}{\sqrt{(2\pi)}\ \sigma_y} \exp\left(-\frac{y^2}{2\sigma_y^2}\right) \tag{9-2}$$

in Equation 9-1 is replaced by

$$(3\sigma_y)^{-1} \quad \text{for} \quad -1.5\sigma_y \le y \le 1.5\sigma_y \tag{9-3}$$

The method has been refined by the authors of the German Risk Study, who used a cross-plume profile with four distinct steps (Aldrich, Bayer, and Schueckler, 1979), which is therefore more nearly akin to the true Gaussian shape.

9.3.1.2 The Dispersion Parameters $\sigma_z(x)$ and $\sigma_y(x)$: Stability Categories

The quantities $\sigma_z(x)$ and $\sigma_y(x)$ have been adjusted by various authors to make the Gaussian distribution fit the measured data. The many possible parametrizations have been reviewed by Gifford (1976). In general, there are two important considerations: the dependence of σ_z and σ_y on the degree of atmospheric instability (or equivalently the intensity of turbulence in the boundary layer) and whether the sigmas are best described as functions of travel time t or of travel distance x. Since most consequence-modeling codes require the user to input stability categories that he has to define, it is worth going over their definition in some detail.

A widely used system for turbulent-diffusion typing was originally proposed by Pasquill (1961), who presented information on the lateral spreading ϕ and the vertical spreading H of diffusing plumes. The latter was shown as a graph and the former as a table. Both are functions of six atmospheric-stability classes, A through F, varying from the "extremely unstable" category A--that is, rapid diffusion--to the "stable" category F, with relatively slow diffusion. The stability category is chosen by reference to a table (see Table 9-2) that defines these categories in terms of the observed wind speed, cloud cover, and insolation conditions--quantities that are widely and routinely observed throughout the world. The values of H and ϕ can be converted into families of curves of the plume standard deviations σ_z and σ_y (Gifford, 1961).

Pasquill's stability categories were chosen subjectively (Gifford, 1976); however, they are approximately linearly related to the intensity of turbulence (Luna and Church, 1974). Ideally, the definition of stability categories should be based on quantities directly related to

Table 9-2. Meteorological conditions defining Pasquill turbulence types

Surface wind speed (m/sec)	Daytime insolation			Nighttime cloudiness[a]	
	Strong	Moderate	Slight	$\geq 4/8$	$\leq 3/8$
<2	A	A-B	B	--	--
2	A-B	B	C	E	F
4	B	B-C	C	D	E
6	C	C-D	D	D	D
6	C	D	D	D	D

[a]The fraction of the sky covered by clouds.

turbulence intensity. Such a quantity is σ_θ, standard deviation of a horizontal wind-direction trace (Singer and Smith, 1966). Indeed, the NRC, in Regulatory Guide 1.23 (USAEC, 1972), has recommended the use of bands of σ_θ for defining stability categories, and these are displayed in Table 9-3. However, a workshop held by the American Meteorological Society (AMS, 1977) did not recommend the use of σ_θ as the basis for determining the vertical standard deviation σ_z. It appears that the method needs further refinement before it can be easily applied (Sedefian and Bennett, 1980).

Another typing scheme recommended by the NRC in Regulatory Guide 1.23 is the ΔT method, which was also used in the Reactor Safety Study. It directly relates the stability category to the value of the atmospheric temperature gradient dT/dz, as shown in Table 9-3. This is an attractive scheme from the user's point of view because, in general, the values of dT/dz can easily be estimated from measurements of the temperature difference ΔT between two points on the meteorological tower. The reliability of the ΔT method has been questioned by several authors (Weber et al., 1977; Sedefian and Bennett, 1980). Vogt et al. (1978) have proposed a method for determining turbulence regimes on the basis of both wind speed and ΔT, parameters that are usually available at reactor sites. Table 9-4 gives an example of such a scheme, developed at Sandia National Laboratories for use in consequence models. Appendix D1 gives reasons why the scheme in Table 9-4 is recommended for consequence modelers. The paper by Gifford (1976) is recommended as a comprehensive review of turbulent-diffusion typing schemes.

9.3.1.3 Parametrizations of σ_z and σ_y

In the Pasquill-Gifford scheme, the parameters σ_z and σ_y are generally presented as functions of travel distance x. Turner (1969) and Doury (1972, 1976), however, use parametrizations that depend on travel time t. There is little to be said here other than that the existence of widely accepted schemes like those of Pasquill, Turner, and Doury, which use such

Table 9-3. Ranges of values of σ_θ and ΔT corresponding to the Pasquill-Gifford stability categories

Stability category	σ_θ (10 m) (degrees)	ΔT (K/100 m)
A	>22.5	<-1.9
B	17.5 to 22.5	-1.9 to -1.7
C	12.5 to 17.5	-1.7 to -1.5
D	7.5 to 12.5	-1.5 to -0.5
E	3.75 to 7.5	-0.5 to 1.5
F	2.0 to 3.75	1.5 to 4.0
G	<2.0	>4.0

Table 9-4. Stability categories defined by reference to both temperature difference and wind speed[a,b]

Wind speed (m/sec)	<-1.9	-1.9 to -1.7	-1.7 to -1.5	-1.5 to -0.5	-0.5 to 1.5	1.5 to 4
	ΔT/ΔZ (°C/100 m)					
	Stability as defined by NRC Regulatory Guide 1.23					
	A	B	C	D	E	F
<2	A	B	B	B	E	F
2 to 3	A	B	C	C	E	F
3 to 5	B	B	C	D	E	F
5 to 6	C	C	C	D	D	D
>6	C	C	C	D	D	D

[a]Scheme based on the typing scheme recommended by Pasquill and the scheme presented in NRC Regulatory Guide 1.23 (USAEC, 1972).

[b]From D. J. Alpert and D. C. Aldrich, Sandia National Laboratories, "Note on Turbulence-Typing Schemes for Use in Reactor Accident Consequence Models," to be published.

different parametrizations of σ_z and σ_y, "probably fairly well reflects the uncertainty of the data" (Gifford, 1976).

In general, consequence modelers use schemes with six or seven stability categories. An example of a widely used scheme appears in Figure 9-5, and a parametrization suitable for use on a computer has been given by Hosker (1974). This model has the advantage that dependence on surface roughness is incorporated. In the Reactor Safety Study, the parametrizations of σ_y and σ_z are a version of the Pasquill-Gifford scheme attributed to Martin and Tikvaart (1968) and described by Eimutis and Koricek (1972). Vogt et al. (1978) have developed a set of parametrizations resulting from tracer measurements over a terrain of major surface roughness. Again, the paper by Gifford (1976) is excellent background reading in this area.

Several of the available parametrizations of σ_y and σ_z were compared in the Benchmark exercise; included were those used in the Reactor Safety Study and the German Risk Study, the schemes by Hosker and Doury alluded to above, and others used in the United States, Europe, and Japan. It appears that, for releases near ground level, the predicted values of χ for a release lasting 1 hour can vary by an order of magnitude simply through the choice of σ_y and σ_z. The reasons for these differences are to be fully discussed in the Benchmark document (see footnote on page 9-17).

For releases of short duration, the predicted time-integrated concentration in the plume is likely to be within a factor of 3 of the actual concentration if measured values are used for all parameters and the correct stability has been assigned. The values of the parameters in the models are most reliable for dispersion over distances of up to a few tens of kilometers; when considering dispersion over distances approaching 100 km, predictions are likely to be increasingly less accurate (Clarke

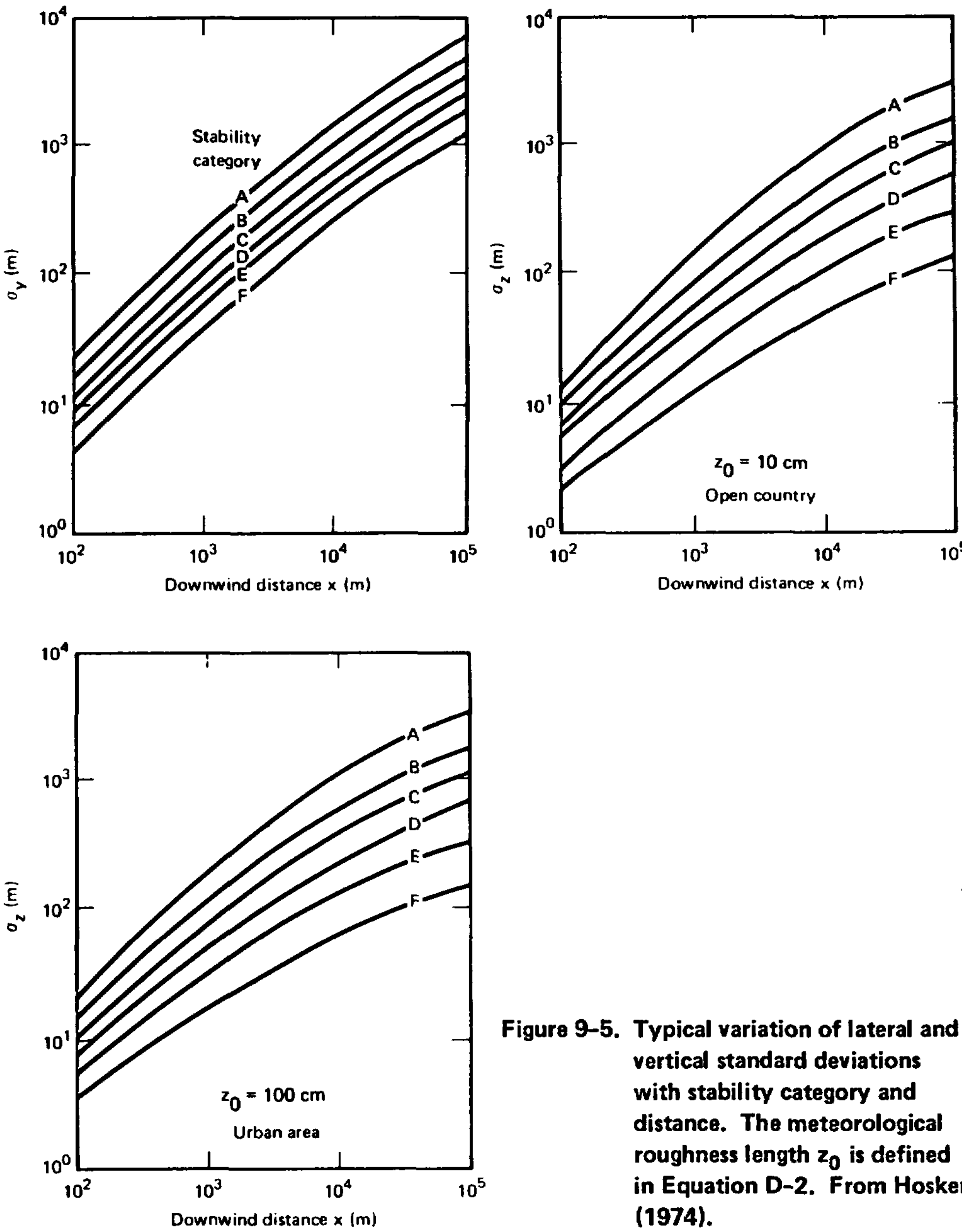

Figure 9-5. Typical variation of lateral and vertical standard deviations with stability category and distance. The meteorological roughness length z_0 is defined in Equation D-2. From Hosker (1974).

et al., 1979). The validity of the Gaussian approach is discussed more fully in Appendix D1.2.

9.3.1.4 Very Low Wind Speeds

Equation 9-1 clearly breaks down as the wind speed tends to zero, and the user of consequence models needs to know what to do when his meteorological data indicate a calm. In CRAC and CRAC2, the code automatically assigns a wind speed of 0.5 m/sec if the true wind speed is lower. In this case, the stability category remains as indicated. Some experiments have been carried out at low wind speeds (Sagendorf, 1974; Gifford, 1976). In all cases the observed diffusion was very much greater than that predicted by the application of the Gaussian model with the appropriate stability category. One possible objection to the neglect or approximation of low-wind-speed conditions--that ground-level concentrations may be very much greater than observed even under category F or G conditions and that the worst case is being ignored--can therefore be discounted.

9.3.1.5 Specific Effects

The basic Gaussian model must be modified to take account of various effects that cannot be neglected. The most important of these are dry and wet deposition, which have been allocated a section of their own (Section 9.3.2). The specific effects covered in this section are radioactive decay, duration of release, building wakes, inversion lid, and plume rise.

Radioactive Decay

As is well known, a single radionuclide will decay, so that, if there was a quantity Q present at t = 0, a quantity $Q \exp(-\lambda t)$ will remain at time t. The time t can be equated to $x/\bar{u}$ if a constant wind speed is assumed. If there is a chain of radionuclides, the buildup and decay of daughters can be treated by standard methods; a convenient reference is a report by Kaiser (1976).

Duration of Release

If a release is of prolonged duration T_r and the wind direction is nominally unchanging during that period, the action of large-scale eddies will cause the time-averaged plume to be wider than it would be for a release of shorter duration. Naden and Leeds (1972) have described how in principle plume models can be modified to account for long averaging times, but their methods are too cumbersome for general use, and simplifying assumptions are needed. A comprehensive review of the T_r dependence of $\sigma_y(x)$ has been given by Griffiths (1977). A report by Clarke et al. (1979) is a useful reference.

One of the simplest methods is to make the substitution

$$\sigma_y \rightarrow \sigma_y \left(\frac{T_r}{T_E}\right)^p \tag{9-4}$$

where T_E is the duration of release in the experiments from which the values of σ_y were derived. In the Reactor Safety Study, p was taken to be 1/3 and T_E to be 1/2 hour. In CRAC2, p is taken to be 0.2 for 3 minutes $\leq T_r \leq$ 1 hour and 0.25 for 1 hour $< T_r$. A practical upper limit on T_r is 10 hours; T_E is taken to be 3 minutes. This CRAC2 scheme is in accord with the recommendations of the American Meteorological Society workshop (AMS, 1977) and should therefore be preferred. Another method is to break the release into puffs of short duration and to superpose the time-integrated concentrations from each puff. UFOMOD, the code used for the

German Risk Study, incorporates this option (Schueckler and Vogt, 1981). In CRACIT the option of a multiphased release is also available.

Building Wakes

It is very likely that in an accident the radioactive effluent will be emitted into the turbulent reactor-building wake. Unfortunately, as Gifford (1976) has remarked, little is known about the properties of diffusion in the wakes that exist in the atmosphere downwind of the structure, and therefore arbitrary assumptions about the effect of such a wake are required. For example, the quantity $[\pi \bar{u}\ \sigma_y(x)\ \sigma_z(x)]^{-1}$ in Equation 9-1 can be replaced by

$$[\pi\sigma_y(x)\ \sigma_z(x) + cA]^{-1}\ \bar{u}^{-1}$$

where A is the area of the building projected onto a plane perpendicular to the wind direction and c is a constant with a value ≤ 0.5 (Gifford, 1976). Equation 9-5 is a little difficult to manage in the sense that some assumption about concentration profiles within the building wake is also needed, and this can lead to difficulties with the conservation of mass or of released activity.

In CRAC and CRAC2 it is assumed that the concentration profiles are Gaussian both laterally and vertically, with boundaries at the width W or height H of area A. As a result, the airborne time-integrated concentration within the wake is

$$\chi(x,y,z) = \frac{Q\ \exp[(-z^2/2\sigma_H^2) - (y^2/2\sigma_W^2)]}{\pi\sigma_H\sigma_W\bar{u}} \tag{9-6}$$

where $H = 2.14\sigma_H$ and $W = 3\sigma_W$. Equation 9-6 is roughly equivalent to Equation 9-5 with $c = 0.4$.

As a crude approximation, the wake is supposed to persist for a certain number of building heights N (say five); the subsequent atmospheric dispersion is calculated by assuming that there is an area source at this time-integrated distance downwind, in which case it is easy to show that the airborne concentration beyond the end of the wake is given by

$$\chi(x,y,z) = \frac{Q\ \exp\{[-z^2/2\sigma'^2_z(x)] - [y^2/2\sigma'^2_y(x)]\}}{\pi\ \bar{u}\sigma'_z(x)\ \sigma'_y(x)} \tag{9-7}$$

where

$$\sigma'^2_y(x) = \sigma^2_y(x - NH_W) + \sigma^2_W \tag{9-8}$$

and

$$\sigma'^2_z(x) = \sigma^2_z(x - NH_W) + \sigma^2_H \tag{9-9}$$

Equation 9-6 cannot be expected to reproduce the concentration profiles within the wake, but it has the merit of being a simple approximation. In practice it is convenient to take N = 0. This does not affect the airborne concentrations at points far enough downwind to be of interest in reactor safety studies. Other simple approximations can be found in the literature (see, for example, Slade, 1968).

The transition to a point outside the wake is very easily accomplished by Equation 9-7 without the need for time-consuming integrations. Furthermore, the calculations of radiation doses from the passing cloud are also relatively easy. The method described here can therefore be justified as a simple and economical way of taking into account the initial dilution of the plume by the building wake. An example of recent advances in the prediction of pollution concentrations near buildings is given by Britter et al. (1976). The range of existing models will be reviewed in the expected report of the international Benchmark exercise. An experimental investigation of plumes emitted within a reactor complex has been reported by Start et al. (1977).

Inversion Lid

Equation 9-1 has been written so as to make it plain that reflection at the ground has been assumed. In practice, there is also a limit to the vertical spread of the plume because the atmospheric boundary layer is capped by a very stable layer with a strongly positive temperature gradient. The turbulence intensity is much reduced within such a layer, and indeed the base of such a layer forms an effective barrier to the upward dispersion of a plume. This is known as the inversion lid. Holzworth (1964, 1972) has given estimates of the height ℓ of this lid for the United States. In general, consequence-modeling codes assume values of ℓ that are typical of the weather being considered. These values can be obtained from reviews like that of Holzworth.

One simple way of taking the lid into account is to assume multiple reflections at the ground and the lid. If this is done, Equation 9-1 is replaced by

$$\chi(x,y,z) = \frac{Q \exp\left(-y^2/2\sigma_y^2\right)}{2\pi\bar{u}\sigma_z\sigma_y} S(h,z,\ell) \qquad (9\text{-}10)$$

where

$$S(h,z,\ell) = \exp\left[-\frac{(z-h)^2}{2\sigma_z^2}\right] + \exp\left[-\frac{(z+h)^2}{2\sigma_z^2}\right] + \exp\left[-\frac{(2\ell+z+h)^2}{2\sigma_z^2}\right]$$

$$+ \exp\left[-\frac{(2\ell+z-h)^2}{2\sigma_z^2}\right] + \ldots \qquad (9\text{-}11)$$

Ultimately, the plume becomes spread uniformly between the ground and the lid, in which case

$$\chi(x,y,z) = \frac{Q \exp\left(-y^2/2\sigma_y^2\right)}{\sqrt{(2\pi)}\ \sigma_y \ell \bar{u}} \qquad (9\text{-}12)$$

The transition between Equations 9-10 and 9-12 can be conveniently made when $\sigma_z = \ell$, when the two expressions for χ differ by at most a few percent.

Since the treatment of the inversion lid is somewhat arbitrary, other methods are equally acceptable. For example, in the Reactor Safety Study Equation 9-1 is used at all times: σ_z is allowed to grow until it equals 0.8ℓ.

Recent studies indicate that CCDFs are not sensitive to values of ℓ (Sprung and Church, 1977a). On the other hand, certain quantities, such as the final height of plume rise, can be extremely dependent on ℓ (Kaiser, 1981). An instructive treatment of inversion lids that vary with time and of the special case of lids at coastal sites appears in the discussion of the CRACIT code in the Zion study (Commonwealth Edison Company, 1981).

Plume Rise

The treatment of plume rise is a source of uncertainty in modeling the consequences of large hot releases of radioactive material. At first sight, this may seem surprising. After all, more than 100 plume-rise models have been described in the literature, and there have been extensive reviews, such as those by Briggs (1969, 1975).* Nonetheless, these reviews, comprehensive as they are, do not encompass all of the elements necessary in a plume-rise model for radioactive plumes:

1. Definition of the mode of release.

2. Liftoff--the behavior of a buoyant plume in the turbulent wake of a building.

3. Plume trajectory.

4. Ground-level concentrations under a rising plume.

5. Termination of plume rise.

6. Passive dispersion.

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*G. A. Briggs also discusses this topic in "Plume Rise and Buoyancy Effects," a draft chapter (1979) for Atmospheric Science and Power Production, the projected replacement for Meteorology and Atomic Energy--1968.

These elements are to be thoroughly discussed in the forthcoming report of the international Benchmark exercise. They are also described by Kaiser (1977, 1981) and Fryer and Kaiser (1979, 1980); a more-detailed discussion appears in Appendix D2.

9.3.2 DEPOSITION PROCESSES

9.3.2.1 Dry Deposition

The standard way of dealing with deposition is to assume that, if $\chi(x,y,0)$ is the ground-level time-integrated concentration in curies per second per cubic meter, the deposited activity is given by

$$\chi_D(x,y) = v_d \chi(x,y,0) \quad \text{Ci/m}^2 \tag{9-13}$$

where v_d is the velocity of deposition, which can occur by a number of processes, including gravitational settling, turbulent and molecular diffusion, and inertial impaction (Horst, 1977). Sehmel (1980) has tabulated some 80 factors that influence dry-deposition rates! The concept of deposition velocity is introduced here to help the reader understand the microphysical processes involved in dry removal.

Particulate Matter

For particles, v_d depends on a variety of parameters: the chemical properties of the material being deposited, the size and shape of the particles, the surface-roughness length z_0, the nature of the vegetation, the atmospheric stability category, and so on. As a result, a survey of published data on the value of v_d produces figures varying between 0.0001 and 20 cm/sec (Hosker, 1974). Since this remains an area of great uncertainty, it is discussed in some depth in Appendix D, where it is shown that, for particulate matter emitted in the aftermath of a reactor accident, it is reasonable to expect v_d to be in the range 0.1 to 10 cm/sec. Hence, the value of 1 cm/sec chosen for use in the Reactor Safety Study seems as good as any other. The large range of possible values for v_d has prompted speculation that v_d should be treated probabilistically. It is pertinent to remark in this context that Beyea (1978a,b) incorporates v_d into his models as an uncertain parameter that varies between 0.1 and 10.0 cm/sec for stability classes A through D, 0.1 to 3.0 cm/sec for stability class E, and 0.1 to 1.0 cm/sec for stability class F.

The deposition velocity v_d is one of the parameters to which many of the results of consequence calculations are sensitive since, as can be seen from Figure 9-3, deposition on the ground is the starting point for most of the pathways to people. (See also Section 9.6.4.1.)

Vapors

The important fission products that have been considered to be gases or vapors in past consequence analyses are the noble gases, elemental iodine, and iodine as methyl iodide. Their deposition velocities are discussed in Appendix D, and the following conclusions can be drawn:

1. The deposition velocities for the noble gases should be taken to be zero since the noble gases are almost totally unreactive.

2. There is no need to consider iodine separately from particulate matter. Experimental evidence shows that it is unlikely that iodine will be released from the fuel in its elemental form; it will probably be in the form of a metallic iodide, most probably cesium iodide (Campbell et al., 1981), which is far less volatile than elemental iodine. Hence v_d should be the same as for particulate matter.

3. Methyl iodide can be neglected in consequence calculations because it makes only a small contribution to public risk. This is a lesson that has emerged from experience gained during the Reactor Safety Study.

9.3.2.2 Modification of the Gaussian Formula

The modification of Equation 9-1 to take into account deposition is achieved by replacing Q (the total emitted activity) by Q(x), the activity remaining at a distance x downwind, where

$$\frac{Q(x)}{Q} = \exp\left\{-\left(\frac{2}{\pi}\right)^{1/2} \frac{v_d}{\bar{u}} \int_0^x \frac{dx'}{\sigma_z(x')} \exp\left[-\frac{h^2}{2\sigma_z^2(x')}\right]\right\} \qquad (9\text{-}14)$$

The proof of this result can be found in the article by Van der Hoven (1968) in Meteorology and Atomic Energy. Since, at distances of many tens of kilometers in the more stable weather categories, Q(x) may be less than a tenth of Q, the modification contained in Equation 9-14 must be included in Equation 9-1.

Appendix D discusses the validity of Equation 9-14 and shows that it is adequate in almost all circumstances that are likely to be considered in a typical consequence analysis. Modifications that may be needed in the future to account for gravitational settling are also discussed.

9.3.2.3 Wet Deposition

If a plume of radioactive material encounters rain as it travels downwind, aerosols will be deposited onto the ground. This wet-deposition process, also known as precipitation scavenging, occurs in one or both of two ways. The first is in-cloud scavenging, which takes place because the radioactive aerosol is a source of condensation nuclei that act as centers for the formation of water droplets. This form of precipitation scavenging is known as rainout. The second occurs through rainfall from clouds above the plume. The falling water droplets collide with and collect the particles that make up the radioactive plume; this is called washout. The fraction of material removed from a plume per unit time is known as the washout coefficient and can be defined as

$$\Lambda = -\frac{1}{\chi}\frac{d\chi}{dt} \tag{9-15}$$

The theoretical calculation of Λ is not easy, because a great number of physical processes are involved, examples being thermophoresis, diffusiophoresis, turbulence, raindrop evaporation, and condensation. It is therefore necessary to make simplifying assumptions; for example, the analysis of experimental data suggests that (Ritchie et al., 1976)

$$\Lambda = CR^{\alpha} \tag{9-16}$$

where R is the rainfall rate (mm/hr); α is a constant, taken to be unity; and C is a constant that can be on the order of 10^{-4} sec^{-1}/mm-hr for stable and neutral atmospheric conditions or 10^{-3} sec^{-1}/mm-hr for unstable atmospheric conditions (i.e., a convective storm) (Ritchie et al., 1981b). The quantity Λ can vary from 10^{-5} to 10^{-2} sec^{-1}.

For the simple case in which it rains everywhere at a constant rate, Equation 9-1 becomes

$$\chi(x,y,z) = \frac{Q \exp[-y^2/2\sigma_y^2(x)]}{2\pi\ \sigma_z(x)\ \sigma_y(x)\ \bar{u}}\ S(h,x,z)\ \exp\left(-\frac{\Lambda x}{\bar{u}}\right) \tag{9-17}$$

where

$$S(h,x,z) = \exp\left[-\frac{(z+h)^2}{2\sigma_z^2(x)}\right] + \exp\left[-\frac{(z-h)^2}{2\sigma_z^2(x)}\right]$$

and Λ is chosen by the user. The quantity of material deposited on a unit area of the ground at the point (x,y,0) is

$$\chi_D(x,y) = \chi(x,y,0)\left\{v_d + \Lambda\ \sqrt{(2\pi)}\ \sigma_z(x)\ \exp\left[+\frac{h^2}{2\sigma_z^2(x)}\right]\right\} \tag{9-18}$$

where v_d is the dry-deposition velocity.

In general, R and Λ depend on time and position since a typical rainstorm moves and is highly structured. Ritchie et al. (1976), for example, describe a simplified rainstorm that covers several tens of thousands of square kilometers and may persist for up to a few days. This area, known as the synoptic region, contains regions known as large mesoscale areas (LMSAs), which cover about 4000 km^2 each and take up about one-third of the total storm area. These areas are not fixed but undergo continuous periods of growth or decay with a lifetime of 12 hours or so; the typical rainfall rate inside them is twice the rate outside them. Within each LMSA there are five or six small mesoscale areas, covering typically 250 km^2, in which the storm lasts for about an hour with a rainfall rate four times that in the synoptic region. Finally, each of these small mesoscale areas contains "cells," 10 km^2 or so in area, in which the storm persists for short periods and in which the rainfall rate can be 25 times that in the synoptic region. The model of Ritchie et al. also allows for the phenomenon of runoff, that is, the washing of deposited radioactive material into rivers and lakes by rain. This complex effect is not, to the authors' knowledge, incorporated into any currently available, fully probabilistic consequence-modeling code, however.

In the context of the discussion of rain, it is pertinent to note that, in CRAC2, for example, it is conventional to assume that, once the plume has passed beyond the farthest point of the computational grid, it is assumed to be completely deposited on the ground by the action of rain, in an interval such as that between 500 and 2000 miles. This artificial procedure is implemented in order to avoid a well-known difficulty in consequence modeling, the nonconvergence of the total population dose (in man-rem). In brief, most dispersion models would predict radiation doses decreasing like r^{-a} at large distances r from a reactor, with $a < 2$. Assuming that the plume is confined to a sector of angular width θ, with a uniform population distribution, the whole-body population dose is proportional to

$$\int_0^{\infty} \frac{\theta r \, dr}{r^a} \propto r^{2-a}$$

which does not converge. This is an unrealistic result, because various depletion processes will act on the plume as it moves to very large distances. The washout of the plume described above is an artificial, but reasonable, means of avoiding this difficulty.

An alternative would be to truncate the above integral when the radiation doses become negligible--some small fraction of those delivered by the natural background, such as 10 mrem. Such truncations are always controversial, however, and it is preferable to use the plume-washout method.

9.3.2.4 Changing Weather Conditions

It is clear from the foregoing that a realistic treatment of the effects of rain can be achieved only within a scheme that treats changes in weather conditions over time. The most significant difference between the predictions of workers who use a statistical model and those who rely on methods that do not incorporate changing weather conditions is to be found in the quantity and the position of deposited gamma emitters. If a radioactive plume encounters a region of heavy rain as it passes over a city some distance downwind, a relatively large fraction of the material within it could be deposited in a densely populated area. This means that the predicted dose rates due to irradiation by deposited gamma emitters could be much higher than would ever be predicted by a code like TIRION (Kaiser, 1976), in which the weather conditions are constant. As a result of rainout, people living in the city could rapidly accumulate a dose to the bone marrow that would exceed the thresholds for early death or morbidity. The number of early deaths predicted for an airborne release of radionuclides is extremely sensitive to assumptions about the dose delivered by deposited gamma emitters.

Codes like CRAC incorporate provisions for changing the weather conditions as the plume travels downwind. It is possible to distinguish a hierarchy of codes of increasing levels of sophistication:

1. Constant-weather codes like TIRION (Kaiser, 1976).

2. Codes with changing weather conditions but an unchanging wind direction, an example being CRAC2 (Ritchie et al., 1981a).

3a. Codes with changing weather conditions and wind directions, and single-station meteorological data, an example being UFOMOD (Schueckler and Vogt, 1981).

3b. Codes with changing weather conditions and wind directions, multiple-station meteorological data, and the effect of topographical features; an example is CRACIT (Commonwealth Edison Company, 1981).

The use of codes belonging to one or another of the stages in this hierarchy is an extremely important element in the current debate among consequence modelers about how best to handle changing weather conditions. This extremely important aspect of consequence modeling is discussed in depth in Appendix D.

9.3.3 PROCESSES THAT LEAD TO THE ACCUMULATION OF RADIATION DOSES

There are five processes that account for most of the ways in which people can accumulate a radiation dose after an accidental release of radioactive material to the atmosphere:

1. Inhalation.

2. Exposure to external irradiation from the passing cloud (cloudshine).

3. Exposure to external irradiation from deposited radionuclides (groundshine).

4. Ingestion, including contaminated vegetation, milk, milk products, and crops contaminated by root uptake.

5. Inhalation of resuspended radionuclides.

For estimating early effects, the most important of these pathways are (1) inhalation from the cloud, (2) cloudshine, and (3) short-term exposure from contaminated ground (hours to days). For estimating latent health effects, the important pathways include (1) external exposure from contaminated ground (both short and long term), (2) inhalation exposure from the passing cloud and from the subsequent resuspension of radionuclides, and (3) the ingestion of contaminated foods.

9.3.3.1 Inhalation

The preceding sections have discussed the methods required to calculate the time-integrated concentration $\chi_n^i(x,y,z)$ of the i^{th} member of the n^{th} chain of radionuclides. The total inhaled activity of this nuclide is $I_n^i(x,y)$, given by

$$I_n^i(x,y) = b_r \ \chi_n^i(x,y,z = 0) \quad \text{Ci} \qquad (9\text{-}19)$$

where b_r is the breathing rate, a parameter that depends on the age of the person involved and on his being engaged (or not) in vigorous activity. The breathing rate commonly assumed for adults (ICRP, 1975) is

$$b_r = 2.66 \times 10^{-4} \ m^3/sec \qquad (9\text{-}20)$$

This is the breathing rate averaged over the entire day--that is, 16 hours of light activity and 8 hours of resting:

Activity level	Breathing rate (m^3/sec)
Light (16 hr/day)	3.33×10^{-4}
Resting (8 hr/day)	9.03×10^{-5}
Daily average	2.66×10^{-4}

The breathing rate will clearly vary during different phases of evacuation or sheltering. For example, people preparing to evacuate may well be highly active. People traveling in cars will be somewhere between resting and light activity. People who have retired to their basements to shelter will also most likely be in a light or lesser state of activity. However, these activity levels and associated breathing rates may not account for possible effects of anxiety. In principle, different breathing rates during different phases of the emergency-response procedure should be taken into account.

The calculation of the radiation doses delivered as a result of the inhalation of radioactive material is extensively reviewed in the Reactor Safety Study. The model used there incorporates the ICRP lung model (ICRP, 1966), with a separate treatment for gaseous radionuclides (Bernard and Snyder, 1975). This allows the calculation of a quantity $F_{n,k}^i(t)$, the dose in rem to organ k at time t after the inhalation of 1 Ci of the i^{th}

nuclide of the n^{th} chain, hereafter referred to as nuclide (n,i), at t = 0. Thus the total dose to organ k integrated to time t is

$$D_k(x,y,t) = \sum_n \sum_i F^i_{n,k}(t)\ I^i_n(x,y) \qquad (9\text{-}21)$$

The quantities $F^i_{n,k}(t)$ are known as inhalation-dose-conversion factors.

The library of dose-conversion factors compiled for the Reactor Safety Study was calculated with the code TIMED (Watson et al., 1976). Many of the consequence-modeling codes available in the United States still have the same library. At present, considerable effort is being devoted to the updating of inhalation-dose-conversion factors. Other codes, such as INREM II (Killough et al., 1978a; Dunning et al., 1979; 1981), have been developed. The International Commission on Radiological Protection (ICRP) is making use of a revised version of TIMED (Watson and Ford, 1980) in a systematic update of these dose-conversion factors. Revised guidance was recently published by the ICRP in Publication 30 (ICRP, 1979, 1980). Dose-conversion factors have also been published by the British National Radiological Protection Board (Kelly et al., 1977; Adams et al., 1978; Hunt et al., 1979).

It appears that, in general, these revised dose-conversion factors do not make a significant difference to the results of consequence analyses for LWR plants. This should not be interpreted to mean that the revised factors are numerically similar to those in the Reactor Safety Study. On the contrary, there are some significant differences, particularly among the actinides. However, for the typical inventory of radionuclides that is predicted to escape into the atmosphere in the event of an LWR accident, these differences do not propagate significantly into the results.

One of the questions most frequently asked about dose-conversion factors is whether the age distribution of the population has been properly accounted for since the dose-conversion factors depend on the age of the exposed person. In the Reactor Safety Study, the dose-conversion factors were developed strictly for the adult male.* The assumption of an adult dosimetry model is a convenient simplification. More detailed studies have revealed that the effect of the closer proximity of the organs to each other in an infant or child is approximately offset by lower intake and higher metabolism (Snyder, 1975). Infants compose only about 2 percent of the population. Thus, even if the dose factors for children were fivefold greater than adult factors, the error in the collective dose would be only about 10 percent, well within the overall uncertainty. Therefore, the convenient approach of using adult parameters for dose calculations does not in general cause significant errors for LWR-accident consequence calculations involving the whole population.

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*The same remark applies to the dose-conversion factors for ingestion, cloudshine, and groundshine, which are discussed in Sections 9.3.3.2 and 9.3.3.3.

The NRC is funding the development of a comprehensive library of dose-conversion factors, which is to be suitable for easy use by consequence modelers and is to be published in 1983 by Sandia National Laboratories.

9.3.3.2 External Irradiation

External Irradiation from the Passing Plume (Cloudshine)

For estimating external cloudshine exposure, let the time-integrated airborne concentration of nuclide (n,i) be $\chi_n^i(x,y,z)$ and let the radionuclide deliver a radiation dose from exposure to cloudshine, $D_{cn}^i(x,y,z)$, to a mathematical element of tissue at point (x,y,z). If the cloud is assumed to be infinite in extent and of uniform concentration, then

$$D_{cn}^{i}(x,y,z) = C_n^i \chi_n(x,y,z) \qquad (9\text{-}22)$$

where C_n^i is known as the cloud-dose-conversion factor and Equation 9-22 is an expression of the well-known semiinfinite-cloud approximation. In general, Equation 9-22 is evaluated with z set equal to 1 m, that is, for a person standing at ground level. The quantity C_n^i does not take into account self-shielding of the body, a subject that is discussed below. For a mixture of radionuclides, the total radiation dose is obtained by summing Equation 9-22 with respect to n and i.

If the cloud is finite, the semiinfinite approximation is not applicable (Van der Hoven and Gammill, 1969), and C_n^i must be multiplied by the quantity $CF(\sigma_z, z/\sigma_z)$, which is a correction factor to be applied when the cloud has its center at a height z above the ground and a vertical standard deviation σ_z. Table VI 8-1 of the Reactor Safety Study contains a compilation of these cloud-dose-correction factors as a function of σ_z and z/σ_z, taken from Meteorology and Atomic Energy--1968 (Slade, 1968). The finite-cloud dose is calculated as follows: the dose is calculated as if the person were located in a semiinfinite cloud with a uniform concentration equal to that at the centerline of the cloud. The correction factor $CF(\sigma_z, z/\sigma_z)$ accounts for the finite extent of the cloud and the vertical displacement (z) between the cloud centerline and ground level.

It is important to remember that the product $CF(\sigma_z, z/\sigma_z)C_n^i$ is an approximation to a three-dimensional integral over the plume. This integral must in principle be evaluated numerically for each gamma ray emitted by each radionuclide in the atmospheric release of radioactivity. This can be extremely time consuming, and it is often the most costly calculation in a consequence-analysis code. It is therefore highly desirable to approximate the integral by an expression not requiring an integration; hence the need for approximations like $CF(\sigma_z, z/\sigma_z)$. It is recommended that the user of consequence-modeling codes use these time-saving approximations as much as possible.

Thus CRAC2 contains an array of values of $CF(\sigma_z, z/\sigma_z)$ for selected values of σ_z and z/σ_z, between which interpolation is carried out for other values of these parameters. It also contains a library of quantities $C^i_{n,k}$ that are related to C^i_n as follows: the dose delivered to a particular body organ k through external irradiation by gamma rays does not necessarily equal C^i_n since the organ may be shielded by the rest of the body. Hence, CRAC2 contains values of C^i_n modified for each body organ k. The modifications were calculated with the code EXREM III (Trubey and Kaye, 1973).

External Exposure from Gamma Radiation Emitted by Deposited Radionuclides (Groundshine)

The deposited activity of nuclide (n,i) per square meter is given by $\chi^i_{Dn}(x,y)$. At time t after the accident, this quantity will have changed because of the action of two mechanisms. The first is radioactive decay, which can be treated in a standard manner and changes the term $\chi^i_{Dn}(x,y)$ to $\chi^i_{Dn}(x,y,t) = \chi^i_{Dn}(x,y)\ RD^i_n(t)$, where $RD^i_n(t)$ accounts for radioactive decay and daughter buildup over time t. The second is weathering, which reduces the gamma dose observed above a contaminated surface by a variety of mechanisms, including the removal of dust by the wind, the carrying away of material dissolved in water, the penetration of radionuclides into the soil, and uptake by vegetation. Therefore, the concentration of each nuclide should be modified by a weathering factor $f^i_n(t)$ that in principle should be different for each radionuclide. In practice, the only nuclide for which much information is available is Cs-137 (Gale et al., 1964). It has been shown experimentally that, if the dose rate above land contaminated by Cs-137 is $\dot{D}_g(t = 0)$ immediately after the contamination has occurred, the dose rate t_y years later is

$$\dot{D}_g(t_y) = \dot{D}_g(t = 0)\ \exp(-0.023t_y)\ \left[0.63\ \exp(-1.13t_y) + 0.37\ \exp(-0.0075t_y)\right] \tag{9-23}$$

The single exponential $\exp(-0.023t_y)$ gives the rate of radioactive decay for Cs-137. The term in brackets is the weathering factor for this nuclide.

The weathering of other nuclides is discussed in Appendix VI of the Reactor Safety Study (USNRC, 1975). It is concluded that so little is known about this subject that it is as well to assume that all nuclides behave like cesium, apart from the different radioactive-decay constants.

At time t after the initial deposition has taken place, the rate at which nuclide (n,i) delivers a radiation dose through groundshine is

$$\dot{D}^i_{gn}(x,y,t) = \chi^i_{Dn}(x,y)\ RD^i_n(t)\ f^i_n(t)\ \dot{G}^i_n \tag{9-24}$$

where $\dot{G}^i_n$ is the dose rate at a reference height Z_r (usually 1 m) above a surface uniformly contaminated by 1 Ci/m^2 of nuclide (n,i). Methods for calculating $\dot{G}^i_n$ have been described (Slade, 1968). The values of $\dot{G}^i_n$ are used as approximations to a two-dimensional integral over the contaminated area. As with the cloudshine, they have been introduced in order to eliminate the need for time-consuming numerical integration.

A code like CRAC2 contains a compilation of dose-conversion factors for groundshine, $G_n^i(t)$, calculated by integrating $\dot{G}_n^i$ for two time intervals--8 hours and 7 days. The CRAC2 data bank contains these quantities $G_n^i(t)$ for each of several organs k (see Section 9.4.8.2).

9.3.3.3 Ingestion

A thorough discussion of the ingestion-exposure pathways is presented in Appendix VI (Chapter 8 and Appendix E) of the Reactor Safety Study (USNRC, 1975). The paragraphs that follow lean heavily on that discussion.

There are two distinct periods of ingestion hazard. Immediately after deposition a significant portion of the radioactive material could be deposited on vegetation that is consumed by people or by animals furnishing food for people. Only a single crop would be affected by direct deposition, so that the potential for exposure would exist for less than a year. (This is the only significant mechanism for ingesting the short-half-life radionuclides like I-131.) The level of contamination on the vegetation would decrease with time because of the influence of weather; for example, wind and rain would remove deposited material from vegetation.

The radioactive material deposited on the soil would be available for incorporation into vegetation by uptake through the roots. This is a long-term exposure mechanism and is relatively unimportant in comparison with the others discussed above. The radioactive material contaminating the soil would be available for plant uptake over a period of several years, but generally only a few percent, at most, would be taken up by plants in one growing season. With time, the material may become unavailable for uptake by plants by migrating below the root zone, for example.

The metabolic characteristics of the radionuclides in people and animals determine which of them would contribute significantly to the "internal" dose. These radionuclides have been identified in extensive experimental studies of fallout from nuclear weapons. The radionuclides selected in the Reactor Safety Study were I-131, I-133, Sr-89, Sr-90, Cs-134, Cs-136, and Cs-137. The radioiodines were considered only for the ingestion of milk because of their short half-lives. It should be noted, however, that chronic exposure may be highly dependent on agricultural practices and food-consumption patterns, and the relative importance of certain radionuclides may be changed. These practices and patterns should be considered carefully in any site-specific application of a consequence-modeling code before assuming that the treatment contained in the Reactor Safety Study is applicable.

Direct Contamination of Vegetation

The calculation of contamination levels on vegetation involves a large number of parameters, many of which are poorly known or extremely variable. There can be large variations in local conditions that directly affect the level of contamination ingested, but since the areas affected are large, this variability is expected to average out in such a way that the effects of local "hot spots" would be offset by a person's consuming food from a wide area.

For the specific reactor site and date of accident, a test should be made to determine whether the accident occurs during the growing season for crops or forage. If not, then the direct contamination of vegetation is not considered to be a feasible mode of radiation exposure.

The major factors considered in calculating the ingestion of radionuclides deposited on vegetation are (1) the fraction of deposited material initially retained on vegetation, (2) its behavior on vegetation as a function of time, and (3) the possible mechanisms that would lead to eventual ingestion by people. The explicit models and data are described in Appendix E to Appendix VI of the Reactor Safety Study, and only a brief discussion is given here.

The fraction of deposited material initially retained on vegetation is taken to be 0.5. Weathering effects would reduce the amount of material remaining on vegetation. The fraction remaining t days after deposition is described by the empirical function

$$f_w(t) = 0.85 \exp\left(- \frac{0.693t}{14.0}\right) + 0.15 \qquad (9\text{-}25)$$

In addition to weathering, radioactive decay would also reduce the amount of radioactivity remaining on vegetation, and this can be treated in a standard way.

The above factors are then used to determine the time required for vegetation-contamination levels to fall to an acceptable level. In the Reactor Safety Study, the criteria by which the acceptability of the levels of contamination can be determined were adapted from recommendations by the British Medical Research Council (MRC, 1975) and the U.S. Federal Radiation Council (FRC, 1964, 1965). For example, the limits set for the milk pathway were 3.3 rem to the bone marrow in the first year from strontium, 3.3 rem to the whole body from cesium, and 10.0 rem to the thyroid from iodine. These radiation doses were related to levels of contamination on vegetation by a model that includes--

1. The initial daily intake of a given radionuclide by an average cow.

2. The decay and weathering processes discussed above.

3. The fraction A' of the activity ingested by the cow that is transferred to the milk. This fraction depends on many factors, including the breed of cow, milk yield, and season.

4. Radioactive decay between the production and the consumption of milk (an average delay of 3 days is assumed).

5. The amount of milk consumed by a person each day, a typical value being 0.7 liter.

6. The ingestion factor, which is the radiation dose delivered to a given organ after the ingestion of 1 Ci of a given radionuclide. Thus, the ingestion factors are similar to the inhalation factors discussed earlier. Examples have been given by Adams et al. (1978).

Factors 1 through 5 above are multiplicative and lead to a concentration factor, which relates deposited radioactivity (Ci/m^2) to the curies ingested by an individual. Consequence-modeling codes generally contain estimates of these concentration factors in a data bank. It should be apparent that these simple factors incorporate many assumptions and complex calculations, and indeed their values are uncertain. Examples of both concentration factors and ingestion factors are given in Table 9-5.

Table 9-5. Examples of parameters used in calculating the dose commitment from ingesting contaminated milk

Nuclide	Concentration factor (Ci/Ci-m^{-2})	Ingestion factor (rem/Ci ingested)[a]		
		Thyroid	Whole body	Bone marrow
I-131	0.692	1.68×10^6	8.79×10^2	2.87×10^2
I-133	0.0042	3.21×10^5	2.70×10^2	1.48×10^2
Sr-89	0.402	5.81×10^2	1.91×10^3	5.26×10^3
Sr-90	0.588	3.18×10^3	5.52×10^4	2.08×10^5
Cs-134	4.22	7.33×10^4	7.14×10^4	7.34×10^4
Cs-136	1.42	9.23×10^3	8.96×10^3	9.29×10^3
Cs-137	4.22	5.55×10^4	5.49×10^4	5.61×10^4

[a]These parameters are based on the "reference man" (ICRP, 1966).

From Table 9-5 it is apparent that the deposited activity of a given nuclide and the subsequent predicted radiation dose accumulated in a given organ are related by the product of a dose-conversion factor for an ingestion concentration factor. Hence a limit like 10.0 rem to the thyroid can readily be translated into an acceptable deposited level of activity for I-131. This level can then be used as the basis for computing areas within which interdictive measures, such as the destruction of crops, are required and the time for which they are needed. The vegetation-contamination models in CRAC consider total radionuclide ingestion from milk, milk products, meat, vegetables, and other foods.

Incorporation of Contaminants from Soil into Vegetation

It is not necessary to calculate acceptable soil-contamination levels for the growing of crops. The uptake of radionuclides by plant roots would be an inefficient mechanism of radiation exposure. At most, a few percent of the deposited radionuclides would be taken up by plants in one growing season. Furthermore, the fraction of material taken up declines rather rapidly in subsequent growing seasons. An area that had enough soil contamination to produce unacceptable doses by plant uptake and the ingestion of food plants would most likely be already forbidden by other restrictions (e.g., external irradiation).

The dose commitment for this mode is calculated as for the direct contamination of vegetation--that is, by making use of concentration factors and dose-conversion factors for ingestion. Rather than the initial retention by vegetation and subsequent weathering, the important factors are the rate of uptake by plants and the rate at which availability to plants decreases (e.g., by leaching to below root zones). Apart from these differences, the two methods are conceptually the same.

9.3.3.4 Resuspension

Radionuclides deposited on the ground will be resuspended by the action of the wind. It is conventional to define a resuspension constant $K(t_y)$ (m^{-1}) such that, if the initial deposited concentration of a radionuclide is 1 Ci/m^2, the concentration in the air just above the ground after t_y years is $K(t_y)$ Ci/m^3. Experimental data on the behavior of $K(t_y)$ as a function of time are meager. A suggested form for $K(t_y)$ is (USNRC, 1975)

$$K(t_y) = 10^{-5} \exp(-0.67t_y) + 10^{-9} \quad m^{-1} \tag{9-27}$$

Experience shows that, for typical predicted releases of radioactivity from light-water reactors, the inhalation of resuspended radioactive material is relatively unimportant (see Tables 9-6 and 9-7). This conclusion would not necessarily be true if actinide releases from a reprocessing plant were being considered, however. An alternative expression for $K(t_y)$ has been given by Anspaugh et al. (1974). A useful review has been published by Linsley (1979). Lassey (1979) discusses a modification of Equation 9-27 that is appropriate for nonarid climates.

9.3.3.5 Discussion

After the preceding, somewhat lengthy, discussion, it is convenient to put the various pathways into perspective by asking, Which are the most important? There is no single answer to this question, since the importance of each pathway varies with, for example, the composition of the radionuclide release, the weather conditions at the time of the accident, and the consequences considered. However, the discussion below leads to some conclusions that are generally applicable to accidental releases from LWR plants.

Figure 9-6 shows the relative doses delivered to the bone marrow at 0.5 mile from the reactor in a BWR-1 release (one of the release categories in the Reactor Safety Study) in neutral weather conditions without rain; it is this dose that has been found to be the most important cause of early fatalities. As can be seen, the predicted radiation doses from inhalation,

Table 9-6. Contribution of different exposure pathways to latent-cancer fatalities for the PWR-1 release category[a,b]

Pathway	Percentage contribution							
	Leukemia	Lung	Breast	Bone	GI tract[c]	All others[d]	Total	Whole body[e]
External irradiation from cloud	0.1	0.1	0.3	0.1	0.1	0.1	1	1
Inhalation from cloud	0.3	22	0.4	0.1	0.5	0.2	24	5
External ground								
<7 days	2	2	5	0.9	0.7	2	13	18
>7 days	8	5	19	2	2	7	43	64
Inhalation of resuspended contamination	0.2	13	0.2	0.3	0.2	0.2	14	4
Ingestion of contaminated foods	1	0.6	2	0.6	0.5	0.7	5	8
Total	12	43	27	4	4	10	100	100

[a]Data from Wall et al. (1977).

[b]This table does not include latent fatalities from thyroid cancer, which are calculated separately, as discussed in Appendix VI of the Reactor Safety Study (USNRC, 1975).

[c]The gastrointestinal tract includes the stomach and the rest of the alimentary canal.

[d]"All others" denotes all cancers except those specified in the table.

[e]Whole-body values are proportional to the 50-year whole-body population dose commitment (man-rem).

Table 9-7. Contribution of different exposure pathways to latent-cancer fatalities for the PWR-2 release category[a,b]

Pathway	Percentage contribution							
	Leukemia	Lung	Breast	Bone	GI tract[c]	All others[d]	Total	Whole body[e]
External irradiation from cloud	0.2	0.1	0.5	0.1	0.1	0.1	1	1
Inhalation from cloud	0.5	4	0.7	0.2	0.4	0.2	6	3
External ground								
<7 days	3	2	7	0.7	0.9	3	16	16
>7 days	12	8	28	3	4	11	66	68
Inhalation of resuspended contamination	0.2	1	0.2	0.4	0.2	0.1	3	2
Ingestion of contaminated foods	2	1	3	1	1	1	9	10
Total	18	16	39	5	6	16	100	100

[a]Data from Wall et al. (1977).

[b]This table does not include latent fatalities from thyroid cancer, which are calculated separately, as discussed in Appendix VI of the Reactor Safety Study (USNRC, 1975).

[c]The gastrointestinal tract includes the stomach and the rest of the alimentary canal.

[d]"All others" denotes all cancers except those specified in the table.

[e]Whole-body values are proportional to the 50-year whole-body population dose commitment (man-rem).

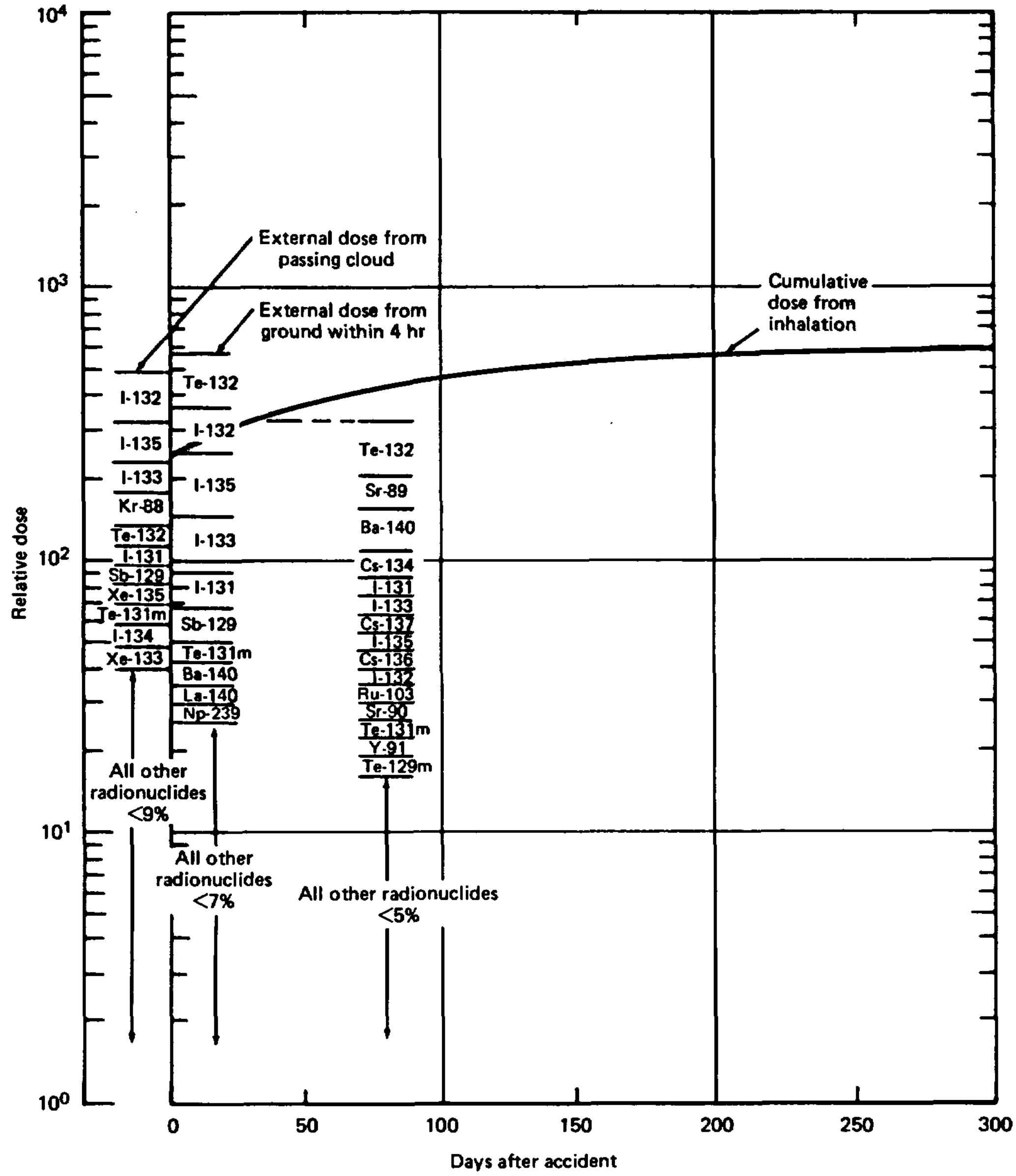

Figure 9-6. Relative doses delivered to the bone marrow at 0.5 mile from reactor. From Wall et al. (1977).

cloudshine, and a 4-hour exposure to groundshine are all comparable, and all of these pathways are important in the calculation of early effects. If the groundshine were extended to 1 day, the relative dose accumulated via this pathway would increase from 500 to about 2000 and would clearly be the dominant contributor (see Figure VI 13-1 of the Reactor Safety Study). This illustrates how important it is to consider evacuation and sheltering strategies that would minimize the effect of this pathway (see Appendix E).

The contribution of different exposure pathways to latent-cancer fatalities is shown in Tables 9-6 and 9-7 for RSS release categories PWR-1 and PWR-2. In both cases, the groundshine radiation dose accumulated over the long term is the dominant contributor, with a further substantial contribution from the external gamma-ray dose delivered in the first 7 days. The long-term external dose from groundshine is dominated by Cs-137 and its daughter Ba-137m. For the PWR-1 release, the inhalation of the cloud and of resuspended nuclides is also important because this release has a high proportion of insoluble, long-lived Ru-106, which preferentially resides in the lung once inhaled. In general, a PWR-2 release is more typical of those expected from an accidental escape of radioactivity from a light-water reactor, since PWR-1 is characteristic of containment failure from a steam explosion, an event that is now thought to be much less likely than was assumed in the Reactor Safety Study.

In calculating the long-term contamination of the ground, leading to the need for relocating people or expensive decontamination, the external dose delivered by gamma rays emitted by deposited Cs-137 is found to be dominant. For estimating the areas within which agricultural produce must be destroyed, the results from the milk pathway are by far the most important.

In summary, the following are generally the most important pathways in calculating the consequences of LWR accidents:

1. Inhalation from the cloud (particularly for early effects).

2. Cloudshine (early effects).

3. Groundshine in the first few hours or days (early effects, latent effects).

4. Groundshine in the long term (latent effects, interdiction of land).

5. Milk ingestion (interdiction of crops).

9.3.4 MEASURES THAT CAN REDUCE PREDICTED RADIATION DOSES

Various protective measures can be envisaged whereby the accumulation of radiation doses by individuals can be much reduced or eliminated. These include evacuation, sheltering, interdiction, and decontamination.

9.3.4.1 Evacuation

It is in the choice of parameters for an evacuation model that the user of consequence-modeling codes can make a highly significant impact on the calculated results. This is illustrated by the CCDF for early fatalities in Figure 9-7, which is taken from Aldrich, Ritchie, and Sprung (1979). The various evacuation schemes used are explained on the figure. It is apparent that the choice of delay time can make a difference of orders of magnitude to the mean public risk.

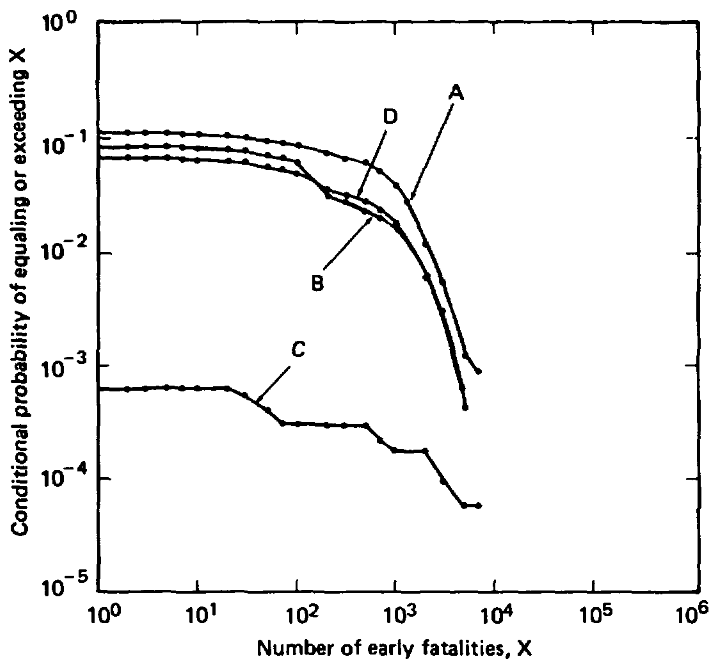

Figure 9-7. Conditional probability versus early fatalities, calculated with the CRAC2 evacuation model. CCDFs are conditional on RSS release categories PWR-1 through PWR-4. Evacuation within 25 miles at a speed of 10 mph. Curves A, B, and C are for 5-, 3-, and 1-hour delays, respectively. Curve D is the weighted sum (5-hour delay, 30%; 3-hour delay, 40%; 1-hour delay, 30%). From Aldrich, Ritchie, and Sprung (1979).

Because of the importance of this topic to the user of consequence-modeling codes, Appendix E has been set aside for a relatively thorough review of evacuation.

The most useful references for background reading are (1) the Reactor Safety Study (USNRC, 1975); (2) the German Risk Study (Gesellschaft fuer Reaktorsicherheit, 1980; EPRI, 1981); (3) a review paper (Aldrich et al., 1978); (4) two reports from Sandia National Laboratories (Aldrich, Blond, and Jones, 1978; Aldrich, McGrath, and Rasmussen, 1978), which describe the updated evacuation model contained in CRAC2; and (5) Section 6 of the Zion PRA (Commonwealth Edison Company, 1981), which describes the evacuation model used in CRACIT and shows how to account for the interaction between an existing road network and wind-shift consequence models.

9.3.4.2 Sheltering

The attenuation of gamma rays by buildings and by surface rugosities has already been mentioned in Section 9.2.1.6. For further information, the reader is referred to Appendix E, which gives guidance about the choice of shielding factors for a typical consequence-modeling code.

9.3.4.3 Interdiction

The process of interdiction would involve the denial of land and its improvements for normal intended use. For example, if the land were contaminated to such an extent that a specified radiation dose would be exceeded over a period of time, use of the land could be prohibited until such time as the radiation dose that an individual would receive over the succeeding period has decreased (through radioactive decay and weathering forces) below the specified criterion. In a decreasing order of impact, interdiction could fall into any of the following categories:

1. Total land and asset interdiction for long periods (more than 10 years).

2. Limited land interdiction (restrictions imposed for a few years).

3. Interdiction of crops.

4. Interdiction of milk consumption.

The criteria for establishing any of these categories of interdiction are based on projected doses to the population. Examples have already been given for the milk pathway in Section 9.3.3.3. Other examples can be found in Table VI 11-6 of the Reactor Safety Study and include, for instance, 25 rem to the whole body from external radiation delivered over a period of 30 years to people living in an urban area and 10 rem delivered over the same period to people living in a lightly populated rural area.* Consequence-modeling codes generally establish the areas within which the given acceptable levels would be exceeded. By assuming that people within those areas would be relocated or that crops would be destroyed, the predicted population dose is reduced and hence the number of predicted health effects is also reduced.

As explained above, the area of interdicted land would decrease with time as the level of contamination decreases. However, decontamination would make it possible to recover some of this land immediately.

9.3.4.4 Decontamination

Decontamination, in the broad sense of the word, is the cleanup and removal of radionuclides. The possible modes of decontamination include the physical removal of the radionuclides, stabilization of the radionuclides in place, and management of the environment. The particular procedure used in a given case would depend on many factors, including (1) the type of surface contaminated, (2) the external environment to which the surface is exposed, (3) the possible hazards to people, (4) the costs, (5) the degree of decontamination that is required, and (6) the consequences of the decontamination operation.

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*In practice, CRAC and CRAC2 make use of the criterion of 25 rem in 30 years for both urban and rural areas.

There is a large body of experimental data on the decontamination of structures, pavements, and land. Most of these data were generated for planning reclamation in the event of a nuclear war. Because of differences in the sizes of contaminant particles and in decontamination criteria, some of these experimental data are not directly applicable to reactor accidents. These problems are discussed more fully in Appendix K to Appendix IV of the Reactor Safety Study (USNRC, 1975).

A measure of the effectiveness of decontamination operations is the decontamination factor (DF), which is defined as the contaminant concentration (in curies per square meter) before decontamination divided by the contaminant concentration after decontamination. Therefore, the larger the DF, the better the decontamination method. For example, a 90-percent removal of contaminants from a surface gives a DF of 10; a 99-percent removal gives a DF of 100.

Typical procedures that can be followed to remove radioactivity are as follows:

1. Hard surfaces (roofs, walls, pavements, etc.)
    a. Replacement of roofing material.
    b. Sandblasting of walls and pavements.
    c. Resurfacing of pavements.

2. Land areas (soil, vegetation, etc.)
    a. Vegetation removal and disposal.
    b. Surface soil removal and burial.
    c. Deep plowing.

The maximum decontamination factor that was considered practical on the basis of the review carried out for the Reactor Safety Study, averaged over large areas, is 20. This limitation is based on the practicality of large-scale decontamination operations, the costs, and the consequences of decontamination operations. The German Risk Study suggests a factor of 10; clearly this is another area of uncertainty.

In CRAC and CRAC2, land that is contaminated to between 1 and 20 times the acceptable level is assumed to be decontaminated just sufficiently to bring it down to that level; land that is more severely contaminated is assumed to have the benefit of a full DF of 20. This then reduces the interdiction time required to allow the weathering and decay processes to decrease the contamination to acceptable levels.

9.3.4.5 Miscellaneous

There are some countermeasures that can in principle be incorporated into consequence models, although this is not always done.

Thyroid Blocking

Potassium iodide or iodate, if ingested in time, reduces the amount of radioiodine that can be taken up by the thyroid. The distribution of potassium iodide or iodate tablets, immediately before or after an accidental release of radioactivity, to the population at hazard would reduce thyroid doses. This possibility is not usually considered in consequence models, because the planning procedures, whereby the prompt distribution of a blocking agent would be possible in the event of an accident, have not been implemented for U.S. reactors. A comprehensive examination of this subject has been made by Aldrich and Blond (1980, 1981), who conclude that "although the effective use of KI could significantly reduce the number of thyroid nodules resulting from a serious accident, it would have no, or only minor, impact on other accident consequences, including immediate deaths or injuries, delayed cancer deaths, and long-term land contamination. Therefore, the availability of KI would provide only a supplemental strategy to be considered along with other protective measures."

Ventilation

If people are assumed to be sheltered from external irradiation by, for example, taking refuge in a basement, then it is conceivable that a significant quantity of radioactive material can be excluded from a structure, either by natural effects or by certain ventilation strategies. This could reduce the amount of inhaled radionuclides (Aldrich and Ericson, 1977) and might reduce the inhalation dose by a factor of 2 (Cohen et al., 1979). The Limerick study (Philadelphia Electric Company, 1981) is an example of a recent risk assessment that takes this effect into account.

Medical Treatment

The effectiveness of medical treatment can readily be incorporated into the dose-response relationship that is used to relate the radiation dose to the probability of some health effect. For example, the Reactor Safety Study (USNRC, 1975) proposes three dose-response relationships for early fatalities, assuming minimal, supportive, and heroic medical treatment. Similarly, it is usually assumed that only 5 to 10 percent or so of thyroid cancers are fatal, and this is easily incorporated into a dose-response relationship.

Respiratory Protection

Recently, a study was carried out to determine what benefit, if any, would result if people covered their faces with sheets, towels, or other crude forms of mask while inhaling air contaminated with radioactive material (Cooper et al., 1981). The study consisted of a series of experiments with various fabrics, aerosols, and vapors, followed by an estimation of the likely efficacy of these fabrics as face masks. A summary of the results appears in Table 9-8. A glance at this table reveals a considerable sensitivity to aerosol-particle diameter, which, as is discussed in Appendix E, is a poorly known quantity for accidental atmospheric releases of radioactivity. Nonetheless, it appears that, of the materials likely to be available in an ordinary house, a wet towel folded into four layers could be quite effective, even for small aerosol-particle diameters, with reduction factors of 5 to more than 100 being feasible. The study does not, however, estimate the radiation dose delivered by gamma rays emitted by fission products trapped in the face mask.

Table 9-8. Estimated penetration through expedient respiratory-protection materials[a,b]

Material	Number of layers	Particle diameter (μm) 0.5	1	5	Molecular iodine (I_2)[c]	Methyl iodide[c]
		DRY MATERIALS				
3M respirator[d] No. 8710	2	0.03	0.004	<0.01		
Sheet	20	0.66	0.64	0.020	1.0	0.6[e]
Shirt	15	0.54	0.59	0.070		
Lower-quality towel	20	0.53	0.41	0.015		
Higher-quality towel	6	0.24	0.13	0.01		0.6[e]
Handkerchief	14	0.61	0.54	0.32		
		WET MATERIALS				
Sheet	6	0.91	0.88	0.22	0.45	0.8[e]
Shirt	6	1.0	0.51	<0.02	0.15[f]	1.0[f]
Higher-quality towel	4	0.20	<0.01	<0.01	0.21	1.0
Handkerchief	2	0.98	0.95	0.37	0.10[f]	

[a]Data from Cooper et al. (1981).
[b]Tests at a pressure drop of 50 Pa (0.2 in. H_2O) and a face velocity of 1.5 cm/sec.
[c]Taken from tests at 1.0 cm/sec, assuming penetration is the product of single-layer penetrations.
[d]Available commercially in single-layer thickness.
[e]Not shown to be statistically different from 1.00.
[f]Wetted with 5-wt% baking-soda solution.

It is not yet common practice to incorporate methods of respiratory protection into emergency plans. However, the distribution of respirators to persons within 10 miles or so of a reactor would be easy, as would special sheltering plans that would instruct people to make emergency respirators out of materials they have in their houses. It follows that, in the future, it may be necessary to consider the effects of respiratory protection in carrying out a consequence analysis.

9.3.5 THE EFFECT OF RADIATION DOSES ON THE HUMAN BODY

After calculating the radiation-dose commitments, it is necessary to consider the adverse health effects that may result in the exposed population. Three kinds of health effects from exposure to accidental releases of radioactive material from a reactor are considered: early and continuing somatic effects, late somatic effects, and genetic effects. Early and continuing somatic effects consist of the early injuries and fatalities that are usually observed after high acute doses ($\geq$50 rads to the whole body) and can occur within days to weeks after exposure. Since these effects are observed only at high doses and high dose rates, they are generally, but not always, limited to persons living in the immediate vicinity of the reactor.

Late somatic effects consist of cancer fatalities and illnesses; they are observed in populations several years to decades after exposure. Finally, changes in the genetic coding of chromosomes can affect the well-being and stability of future populations. Unlike the early and late somatic effects, genetic effects manifest themselves not in the irradiated individuals but in their descendants. Consequence-modeling codes usually have the capability of calculating genetic risk. Whether this is done or not depends on the intended application of the consequence model.

Appendix VI of the RSS (USNRC, 1975) provides the most extensive and complete discussion of early and continuing somatic effects available at present. Section 9.3.5.1 discusses the concepts presented in Appendix VI and some of the uncertainty associated with the estimates of early somatic effects.

Late somatic and genetic effects have been studied by many groups, including the Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR) of the National Academy of Sciences, the United Nations Scientific Committee on Radiological Protection (UNSCEAR), the National Council on Radiation Protection (NCRP), the International Commission on Radiological Protection (ICRP), and the RSS Advisory Group on Health Effects. These groups have made estimates of the risk of adverse health effects from the effects observed in exposed populations. Many of these estimates are based on effects from high doses and high dose rates, with extrapolation to the low dose regions. At low doses and low dose rates, it is difficult to distinguish between health effects that result from radiation exposure and those that result from other causes. Many uncertainties are associated with these risk estimates, and some of these are addressed in Section 9.3.5.1.

The health-effects model described in Appendix VI of the RSS is the most commonly used model in codes like CRAC, CRAC2, and CRACIT. It is currently being reviewed by a group of scientists at Harvard University's School of Public Health, under the sponsorship of Sandia National Laboratories. A report by D. W. Cooper and co-workers, detailing the strengths and weaknesses of the model and areas needing further research, is expected to be published in 1983.

9.3.5.1 Early and Continuing Somatic Effects

After a reactor accident, the large doses required to produce early effects could result from several pathways: external irradiation from the passing cloud, external irradiation from ground contamination, and internal irradiation from inhaled radionuclides (see Section 9.3.3.5).

A set of criteria (see Section 9.4.8.3) for relating the dose received by individuals to the early fatalities and injuries that may arise within a year after an accident are detailed in RSS Appendix VI. Early fatalities were estimated by considering damage to the bone marrow, the lung, and the gastrointestinal (GI) tract. Early injuries include respiratory impairment, temporary changes in the GI tract, hypothyroidism, thyroiditis, temporary sterility, congenital malformations and growth retardations, cataracts, and prodromal vomiting. Early injuries are defined by the RSS as the responses that require medical attention. A convenient measure of early injuries is sometimes taken to be the number of people who receive more than 200 rem to the bone marrow or the whole body. These people would require hospital treatment. The consequence modeler should be aware that codes like CRAC2 base most of their illness estimates on impairments requiring medical attention. However, CRAC2 has an option available for considering numerous injuries in detail, if the user finds this type of analysis necessary.

Fatalities

Radiation doses to the bone marrow, the lung, and the GI tract would be the major contributors to the risk of early fatalities. These three organs should be treated on a conditional risk basis, to prevent overestimation. The probability of death from bone-marrow irradiation usually dominates the corresponding probabilities for the lung and the GI tract for LWR consequence calculations.

If an accident were to occur, it is presumed that the emergency response would include medical treatment to mitigate the adverse consequences that may result from high-dose exposures. The RSS established 60-day median lethal doses, $LD_{50/60}$ (i.e., the doses that would be lethal to 50 percent of the exposed population within 60 days), for varying degrees of medical treatment (minimal, supportive, and heroic) for bone-marrow exposures. An $LD_{50/60}$ of 340 rads was recommended by the RSS Advisory Group on Health Effects as the value to use in estimating health effects if only minimal medical treatment is available. With heroic treatment (e.g., bone-marrow transplants) the $LD_{50/60}$ value may be significantly increased, but these medical procedures may have adverse side effects that could decrease the survival rate. For supportive medical treatment, the RSS used an $LD_{50/60}$ of 510 rads (curve B of Figure VI 9-1 of the RSS). The consequence modeler should be aware that the fatality-risk estimates are extremely sensitive to the $LD_{50/60}$ value.

Considering the current medical expertise and future advances, it is reasonable to assume that supportive treatment would be available to persons exposed to high doses after a reactor accident. However, the availability of supportive treatment would depend on the number of people needing hospital treatment for high radiation exposure (more than about 200 rads to the bone marrow). The RSS estimated that, in the United States, 2500 to 5000 people could be given supportive treatment (based on 1975 medical facilities). The consequence analyst should be aware that if the number of individuals receiving acute bone-marrow doses (>200 rads) exceeds 5000, it is likely that some individuals would receive less than supportive treatment (e.g., minimal treatment).

Estimates of lung damage from inhaled beta-emitting radionuclides were adequately covered in the RSS. Other useful references for fatalities from inhaled radionuclides are reports by Filipy et al. (1980) and Hahn (1979).

In the RSS, the dose-response relationship for early fatalities is applied to a radiation dose that is the sum of the following:

1. External dose from the passing cloud.

2. External dose from contaminated ground (the duration of exposure to gamma rays emitted by deposited fission products, together with the degree of shielding, depends on the assumed emergency-response strategy).

3. Internal dose received during the first 7 days from inhaled radionuclides.

4. For bone-marrow exposure only, half of the internal dose from inhaled radionuclides received from day 8 through day 30.

5. For lung exposure only, the internal dose from inhaled radionuclides received from day 8 through day 365.

It can be seen that this is a specifically defined dose commitment. The consequence modeler should be cautioned that redefining the dosimetry assumptions used in the analysis would require redefinition of the dose-response relationships for early fatalities. Repair mechanisms may modify the effects of radiation exposure if the exposure is received over an extended period of time.

Injuries

The various types of impairment listed at the beginning of this section are detailed in Appendix VI of the RSS. A sublethal dose, defined as the dose expected to cause a clinical response in 10, 50, or 90 percent of the exposed population, was estimated for the various morbidities. These responses are not as easily determined as fatalities; thus the estimates have some subjectivity and increased uncertainty.

9.3.5.2 Late Somatic Effects

Late somatic effects consist of latent-cancer fatalities, nonfatal cancers, illnesses, and benign thyroid nodules. The RSS model included a latent period during which there was no increase in cancers and a plateau period with a uniform cancer rate for each cancer type.

The estimates of latent cancer calculated by the CRAC code are based on the BEIR I report (NAS-NRC, 1972), with the leukemia and bone-cancer values modified to reflect new data that became available between 1972 and 1975. The RSS developed three estimates of risk. The upper-bound estimate used the linear, no-threshold estimators from the BEIR I report (1972). The central estimate (see Section 9.4.8.4) incorporated a dose-effectiveness factor for exposures delivered at low dose rates. The lower-bound estimate took into account the large uncertainty in estimating effects from low doses and low dose rates and assumed a threshold of 10 or 25 rem for latent-cancer fatalities. The central-estimate approach is consistent with the BEIR III report (NAS-NRC, 1980), which used a linear-quadratic model to calculate risk estimators for latent-cancer fatalities. In addition, the BEIR III report published ranges that indicate some of the uncertainty associated with these factors. The upper and the lower bounds of the ranges were obtained with the linear model and the pure quadratic model, respectively. The risk estimates, based on the linear-quadratic model, of BEIR III (1980) are approximately 2 times lower than the BEIR I (1972) estimates based on the linear model.

Recently, Loewe and Mendelsohn (1980) conducted some preliminary reassessments of the dose data for people exposed by the atomic bombs at Nagasaki and Hiroshima. Since the BEIR estimates were calculated from these Japanese data, these reassessments could have some impact on the final estimates of latent-cancer risk. The Los Alamos National Laboratory is attempting to redefine the source term from the two bombs. In conjunction with this effort, the Oak Ridge National Laboratory is recalculating dose estimates. Final resolution of the health-effects estimate will likely follow these efforts. It is important that the consequence modeler be aware of these developments.

Except for leukemia, the latent-cancer fatalities presented in Table VI 9-4 of the RSS were calculated for a 30-year plateau period, whereas the BEIR I report (1972) used the remaining lifetime as the plateau period for "solid tumors." A comparison of the values obtained by assuming lifetime and 30-year plateaus is given in Table 9-9. (The lifetime plateau is implemented in the CRAC2 code.)

Table 9-9. Expected latent-cancer deaths per 10^6 man-rem of external exposure

	Expected deaths per 10^6 man-rem	
Type of cancer	CRAC health-effects model[a]	CRAC2 health-effects model[b]
Leukemia	28.4	28.4
Lung	22.2	27.5
Stomach	10.2	12.7
Alimentary canal	3.4	4.2
Pancreas	3.4	4.2
Breast	25.6	31.7
Bone	6.9	10.1
Other	21.6	28.0

[a]Based on a 30-year plateau period for all cancers except leukemia.

[b]Based on a lifetime plateau period for all cancers except leukemia.

In calculating the incidence of thyroid nodules, both benign and cancerous, the RSS assumed the following dose-response relationship, with allowance for the age distribution of the population:

Dose range (rem)	Expected thyroid nodules per 10^6 man-rem: Benign	Cancerous
<1500	200	134
1500-3000	100	67
>5000	0	0

It is assumed that the thyroid gland is ablated after doses higher than 5000 rem, requiring its surgical removal and the daily use of substitute hormone pills. Aldrich and Blond (1980), in a study of the effectiveness of potassium iodide as an emergency protective measure in the aftermath of nuclear reactor accidents, used a simplified model: the probability of a thyroid nodule is 3.34×10^{-4} per rem for doses not exceeding 3000 rem; above 3000 rem, the thyroid is assumed to be ablated.

In the RSS, the radiation dose to which the dose-response relationship is applied is given by the sum of (1) the external thyroid dose from the passing cloud, (2) the external thyroid dose from contaminated ground, (3) the internal dose delivered during the first 30 days by all inhaled radionuclides except iodine-131, and (4) one-tenth of the internal dose delivered during the first 30 days by iodine-131. The RSS offers clinical evidence for the assumption that iodine-131 irradiation causes a lower incidence of thyroid nodules than do external gamma rays. In addition, from a purely radiobiological standpoint, it is thought that the more uniform distribution of dose within the thyroid from external irradiation might increase the induction of clinical hypothyroidism.

The dose-effectiveness factor of 0.1 for iodine-131 was disputed by the American Physical Society Study Group on Light-Water Reactor Safety (APS, 1975), which assumed a range of factors from 0.3 to 1.0. This issue remains unresolved (Aldrich and Blond, 1980); however, the reader should be aware that, in the case of CRAC and CRAC2, the assumption of one-tenth effectiveness for iodine-131 is built into the code.

Finally, most thyroid cancers are well differentiated, slow growing, and treatable. The mortality rate for thyroid cancers is therefore much lower than that for other forms of cancer. The 10-percent mortality rate assumed in the RSS for malignant thyroid cancers is probably an overestimate.

Genetic Effects

The RSS used the BEIR I report (NAS-NRC, 1972) to prepare tables for potential genetic disorders per 10^6 man-rem of external and internal radiation exposure. These genetic effects include autosomal dominant disorders, multifactorial disorders, disorders due to chromosomal aberrations, and spontaneous abortions. The estimates of genetic effects in the BEIR III report (NAS-NRC, 1980) are based on new data that have become available since 1972, but they are not significantly different from the 1972 estimates. Therefore, it is reasonable to continue using the genetic estimators that are implemented in codes like CRAC. In most consequence analyses, genetic disorders are not part of the final output. The latent cancers tend to dominate the risk estimates of latent effects. However, when executing codes like CRAC, the user does have the option to provide estimates of genetic risk.

9.3.5.3 Discussion

As this chapter was being written and reviewed, it became apparent that the topics of dosimetry and dose-response relationships generate considerable scientific controversy. After detailed discussion, the authors have decided to make the following recommendations. First, the state of the art has not yet "solidified" to the extent that it is possible to recommend unequivocally a replacement for the RSS methods. Hence, the RSS remains the best comprehensive treatment of dosimetry and dose-response relationships in the context of consequence modeling, and its methods remain acceptable. Second, because considerable work has been done since the publication of the RSS, those who wish to try to update the methods are encouraged to do so. However, those who vary from the RSS values should use sources that have been subjected to a peer review, such as the BEIR III report (NAS-NRC, 1980), the UNSCEAR (1977) report, and ICRP Publication 26 (ICRP, 1977). Finally, as already mentioned, studies intended to update the RSS methods are in progress: the NRC is funding work on age- and sex-specific dose-conversion factors at the Oak Ridge National Laboratory, and work on health-effects modeling is under way at Harvard University's School of Public Health. When their results have been published, a comprehensive updating of the RSS methods in codes like CRAC2 will be in order.

9.3.6 ECONOMIC IMPACTS

The economic models of consequence-modeling codes require several cost elements. Thus, for example, the cost of evacuating a person is assigned a given value, and the total cost of evacuation is merely that figure multiplied by the number of people evacuated. The quantity of each agricultural product condemned is calculated and multiplied by the value of a unit quantity of that product. The economic model is thus a simple counting and adding routine in which the key factors are the unit costs of each countermeasure; these are required as input to the code and are discussed further in Section 9.4. An illustrative list of required cost inputs is as follows:

1. Evacuation cost per person.

2. Value of residential, business, and public areas per person.

3. Relocation cost per person.

4. Decontamination cost per acre for farm areas.

5. Decontamination cost per person for residential, business, and public areas.

6. Compensation rate per year for residential, business, and public areas (i.e., fraction of value).

7. Average value of farmland per acre for state, county, or smaller areas.

8. Average annual value of farm sales per acre for state, county, or smaller areas.

9. Miscellaneous information, such as seeding and harvesting month, fraction of land devoted to farming, fraction of farm sales due to dairy production.

Hence, the economic model essentially adds the costs incurred for evacuation, relocation, the interdiction of land use, and the destruction of crops. Note that it does not consider the cost of health effects (this would involve assigning a monetary value to life, which would be extremely controversial). It does not consider any economic multiplier effects (e.g., the effect on employment in one area if a factory in another has to close). There is no attempt to incorporate plant costs, although in the prediction of economic risk the in-plant property damage far exceeds the offsite property damage. As demonstrated by the accident at Three Mile Island, the costs of decontaminating and reconstructing the plant and of replacing power may be several billion dollars even though the offsite consequences were very small indeed.

9.4 INPUT-DATA REQUIREMENTS

As already mentioned, it is in the choice of input data that the user can most influence the outcome of the consequence analysis.

9.4.1 BASIC RADIONUCLIDE DATA

The user of a code like CRAC2 is generally expected to provide the inventory of radionuclides present in the reactor core at the time the accident sequence starts. The inventory used in the RSS was that appropriate for a 3200-MWt Westinghouse PWR. It was calculated for an end-of-cycle equilibrium core with the ORIGEN code (Bell, 1973) and has been used to represent both PWR and BWR cores. Differences in reactor size (i.e., thermal power level) are usually accommodated by a linear scaling of the radionuclide inventories.

Since the writing of the RSS, ORIGEN has been updated (Croff, 1980). Moreover, Sandia National Laboratories, using its own version of ORIGEN,*

*The Sandia version of ORIGEN will be described by D. E. Bennett in a forthcoming report entitled Radionuclide Core Inventories for Standard PWR and BWR Fuel Management Plans.

has calculated equilibrium inventories for a 3412-MWt Westinghouse PWR, a 3578-MWt General Electric BWR, and a 1518-MWt Westinghouse PWR. Inventories of selected radionuclides are shown in Table 9-10 (Ostmeyer, 1981) as multiples of the inventory for the 3412-MWt PWR.

The significant differences between the long-lived-radionuclide inventories for the RSS 3200-MWt PWR and the 3412-MWt PWR are due to the assumption of a 25 percent greater fuel burnup for the latter (26,400 MWd/tonne in the RSS, subsequently increased to 33,000 MWd/tonne). In general, the inventories of the short-lived radionuclides are proportional to the power level. The most significant impact of the greater inventory of long-lived radionuclides, especially Cs-137, is to increase the predicted number of latent-cancer fatalities roughly in proportion to the increase in inventory. For the 3412-MWt PWR, the land-interdiction and decontamination results are approximately 30 percent larger than those of the RSS, while those for the 3578-MWt BWR are approximately 50 percent larger again. Differences for other consequences, such as early fatalities, which are more directly dependent on the inventory of short-lived radionuclides, are smaller.

Inventories most representative of the reactor being studied should be employed for reactor-accident consequence calculations. This would include using BWR inventories, if available, for BWR consequence calculations and using radionuclide inventories representative of the reactor-power level under consideration.

Once an inventory is available, it is necessary to reduce the number of radionuclides to manageable proportions, since ORIGEN considers 246 activation products, 461 fission products, and 82 transuranics. Many of these nuclides are not radioactive, but even so, the total number of radionuclides runs to several hundred. Many of them can be eliminated without significantly affecting the results of the radiation-dose calculations. The elimination process in the RSS was based on a number of factors, such as (1) quantity (curies); (2) release fraction; (3) emitted-radiation type and energy; (4) chemical characteristics; and (5) half-life. It is possible to eliminate many short-lived radionuclides because the shortest interval between the termination of the chain reaction (accident-sequence initiation) and the beginning of the escape of radionuclides to the atmosphere is 30 minutes (for the accident sequences presented in the RSS).

In the RSS, the list of nuclides considered was thus reduced to the 54 shown in Table 9-11. Recent studies have shown that, at least for LWRs, this list is more than adequate. Table 9-11 also contains some examples of nuclides that are important in some of the radiation pathways discussed in Section 9.3, for releases from LWRs only. These are taken from Chapter 13 of Appendix VI of the RSS, which presents many more details.

Finally, the list of radionuclides should be accompanied by standard information on parents and/or daughters in a radioactive-decay chain and the radioactive half-life. Also required in principle is the spectrum of the gamma rays emitted by the decaying radionuclides, in order to calculate cloudshine and groundshine (see Section 9.4.8.2) and the deposition velocity and washout coefficient (Section 9.4.5).

Table 9-10. Inventory of selected radionuclides for various reactors[a]

Nuclide	Half-life (days)	Inventory (Ci)					
		End of cycle 3412-MWt PWR	End of cycle 3200-MWt PWR[b]	End of cycle 3578-MWt BWR	End of cycle 1518-MWt PWR	1/3 cycle 3412-MWt PWR	2/3 cycle 3412-MWt PWR
Kr-85	0.117	6.64×10^5	1.03	1.36	0.44	0.68	0.84
Mo-99	2.8	1.66×10^8	0.94	1.05	0.45	1.02	1.01
Tc-99m	0.25	1.43×10^8	1.00	1.05	0.45	1.03	1.01
Ru-103	39.5	1.25×10^8	0.85	1.06	0.44	0.87	0.96
Ru-105	0.185	8.22×10^7	0.88	1.07	0.43	0.86	0.94
Ru-106	366	2.90×10^7	0.86	1.24	0.42	0.66	0.83
Te-129m	0.34	6.70×10^6	0.79	1.06	0.44	0.88	0.96
Te-131m	1.25	1.28×10^7	1.00	1.07	0.44	0.97	0.98
Te-132	3.25	1.27×10^8	0.92	1.06	0.45	1.00	1.00
Sb-129	0.179	2.72×10^7	1.22	1.06	0.44	0.93	0.97
I-131	8.05	8.74×10^7	0.98	1.06	0.45	0.99	1.00
I-132	0.096	1.29×10^8	0.92	1.05	0.44	0.99	1.00
I-133	0.875	1.84×10^8	0.94	1.05	0.45	1.02	1.01
I-134	0.037	2.02×10^8	0.95	1.05	0.45	1.02	1.01
I-135	0.28	1.73×10^8	0.88	1.06	0.45	1.02	1.01
Cs-134	750	1.26×17^7	0.60	1.20	0.38	0.55	0.76
Cs-136	13.0	3.91×10^6	0.77	1.04	0.41	0.67	0.84
Cs-137	11,000	6.54×10^6	0.72	1.39	0.44	0.67	0.83
Ba-140	12.8	1.68×10^8	0.94	1.05	0.45	1.02	1.01
Ce-144	284	9.15×10^7	0.92	1.14	0.45	0.77	0.90

[a]From Ostmeyer (1981).
[b]The reference PWR for the Reactor Safety Study.

Table 9-11. Radionuclides considered in the Reactor Safety Study consequence analysis[a]

Element	Radionuclide	Element	Radionuclide
Cobalt	Co-58,* Co-60*	Iodine	I-131,[c,g,h,i] I-132,[b,g,h] I-133,[b,g,i] I-134, I-135[b,g,h]
Krypton	Kr-85,* Kr-85m,* Kr-87,* Kr-88[b]		
Rubidium	Rb-86*	Xenon	Xe-133, Xe-135
Strontium	Sr-89,[c] Sr-90,[d,e] Sr-91	Cesium	Cs-134,[c] Cs-136, Cs-137[j]
Yttrium	Y-90,* Y-91	Barium	Ba-140[c]
Zirconium	Zr-95, Zr-97	Lanthanum	La-140
Niobium	Nb-95*	Cerium	Ce-141, Ce-143,* Ce-144[f]
Molybdenum	Mo-99	Praseodymium	Pr-143*
Technetium	Tc-99m*	Neodymium	Nd-147*
Ruthenium	Ru-103, Ru-105,* Ru-106[f]	Neptunium	Np-239
Rhodium	Rh-105*	Plutonium	Pu-238,[e] Pu-239, Pu-240, Pu-241[e]
Tellurium	Te-127,* Te-127m, Te-129,* Te-131m, Te-132[b,c,g]	Americium	Am-241*
		Curium	Cm-242, Cm-244
Antimony	Sb-127, Sb-129		

[a]Applicable to releases from LWRs only. The radionuclides marked with an asterisk are negligible contributors to health effects. The most significant contributors are signaled with superscript letters for the modes or effects listed below.

[b]Cloudshine.
[c]Inhalation (early effects).
[d]Leukemia (inhalation dose).
[e]Bone cancer (inhalation dose).
[f]Lung cancer (inhalation dose).
[g]Groundshine (early effects).
[h]Thyroid dose.
[i]Milk-ingestion pathway.
[j]Long-term groundshine.

9.4.2 SPECIFICATION OF THE SOURCE TERM

It is necessary to obtain data that will have been calculated by the methods described in Chapters 7 and 8. For simplicity, this will be referred to as the specification of the source term. An example of this sort of information, taken from the RSS, appears in Table 9-1, the various entries of which are discussed below.

Some of the discussion that follows goes into greater depth than is required for the straightforward use of codes like CRAC2, but, because of the current rapid developments in this field, it is necessary to be able to talk intelligently about the likely future impact on consequence modeling.

9.4.2.1 Magnitude of Radionuclide Releases to the Atmosphere

The activity of each radionuclide escaping to the atmosphere is calculated by the methods described in Chapter 8. In the RSS, the various radionuclides were classified into eight groups (see Table 9-1), mainly on the basis of their volatility. It is still common practice to classify radionuclides in this way in order to reduce the burden of calculation with codes like CORRAL.

The rate of radionuclide release to the atmosphere is, in fact, time dependent, as described in Chapter 8. Current consequence-modeling codes are generally not able to handle this, but since the next generation of radionuclide-transport codes are likely to predict varying rates of release, it is possible that in the future updated versions of consequence models that can accept this input will be required.

9.4.2.2 Timing

Various times are important input to consequence calculations: the time of release, the duration of release, and the warning time.

The time of release, which is a parameter calculated in the modeling of physical processes, is the interval between the start of the accident and the predicted start of the release of radionuclides to the atmosphere. In some consequence-modeling codes, this interval is used to attenuate the source term by the process of radioactive decay. In general, the interface between radionuclide-transport codes and consequence-modeling codes allows for what is rather an artificial means of accounting for this radioactive decay. Future generations of radionuclide-transport codes may include a better treatment of this effect and require a more sophisticated way of handling input on the part of consequence-modeling codes.

The time of release can generally be obtained from the output of codes like MARCH and CORRAL. For accident sequences in which core melt occurs before containment failure, the time of containment failure is also the time of release. For sequences in which the containment fails first, the time of release can be taken as the time at which the gap release or the melt release from fuel would be predicted to occur.

The duration of release, which is generally calculated from the output of a radionuclide-transport code like CORRAL, could be used to allow for (1) radioactive decay and (2) the broadening of the plume by the action of large-scale turbulent eddies in the atmosphere. In principle, changes in wind direction during this period could also be taken into account (see Appendix D4.2) and used as input to an evacuation model for an area affected by a complicated plume like that shown in Figure D-12.

The warning time is the period between the awareness of an impending core melt and the release of radioactivity. This parameter is important in evacuation models. It should in principle be obtained by comparing the emergency-plan implementing procedures (EPIPs) with the output of a code like MARCH. The EPIPs give criteria for the declaration of a general emergency, on the basis of such quantities as the water level in the core and radiation or pressure readings in the containment. From the MARCH output one can usually deduce when these readings will be reached.

9.4.2.3 The Elevation of Release and the Dimensions of the Release

Data on these parameters are obtained from the modeling of physical processes. The simplest models of the atmospheric dispersion of radionuclides assume a point-source release from a known elevation. If the source is assumed to be of this nature, the elevation of release should be provided.

If the release does not emerge from a well-defined source, or if that source is on the face or the roof of a reactor complex, it is assumed that mixing will occur throughout the reactor-building wake. In this case, the dimensions of the reactor complex are needed (i.e., the height and width of the projection perpendicular to the wind direction).

9.4.2.4 Buoyancy

Plume rise can be important in reducing the predicted values of the mean public risk, particularly for early effects. The minimum of information required is the rate of energy release that accompanies the escaping radionuclides, and this should emerge from the modeling of physical processes as described in Chapter 7. The MARCH code contains an output variable, "QSENS," that, as is explained in an associated comment card, is intended for use in the CRAC plume-rise model. Examples are given in Table 9-1.

Ideally, momentum effects should also be considered. In principle, the orientation, size, and momentum of an escaping jet should influence plume rise. In practice, predictions of these effects are not likely to be feasible in the near future, and, in any case, most consequence-modeling codes cannot handle such information.

9.4.2.5 Particle-Size Distribution

This section has more to do with what may be possible in the future than with what is feasible now. The particle-size distribution can affect, among other factors, the dry-deposition velocity v_d (see Appendix D3.1), the washout coefficient, and the health-physics calculations (e.g., the dose-conversion factors for the inhalation pathway; see Section 9.4.8.1).

Some particle-size distribution must be assumed, either explicitly or implicitly. For example, the RSS compilations of inhalation-dose-conversion factors are for an aerosol with an activity median aerodynamic diameter (AMAD) of 1 μm, distributed lognormally. The range of particle diameters that are implicitly assumed to be compatible with the RSS assumption of a dry-deposition velocity of 1 cm/sec is discussed in Appendix D3.

It is very likely that the considerable amount of research that is under way on radionuclide source terms will produce a mass of information on the size distribution of particles emitted to the atmosphere in a severe reactor accident. It might also provide information on particle composition, since such particles will be agglomerates made up of fuel, structural materials, and fission products. It might show that different nuclides may be preferentially associated with different particle sizes. As discussed in Appendix D3.1, such considerations would imply a need for considerable changes in the deposition model normally used in consequence-modeling codes and would also require considerable extra sophistication in the input to such codes.

In practice, the user of a consequence-modeling code is likely to find that he has little freedom of choice. The health-physics parameters that are generally built into a code, where they reside in a data bank, will have been calculated on the assumption of a given particle-size distribution, and this distribution usually has an AMAD of 1 μm. The deposition velocity and washout coefficients are also input without much consideration of particle diameter. Hence, the user will generally not be required to make an explicit decision about particle sizes.

9.4.2.6 Chemical Properties

The chemical properties of the released radionuclides will influence the subsequent health-physics calculations--for example, the classification of elements into the three inhalation classes D, W, and Y, which represent respiratory-clearance half-times on the order of days, weeks, and years, respectively. It is therefore necessary to obtain either a direct classification of elements as D, W, or Y or enough information about their chemical properties for a classification to be made. Again, in practice, the user may well find that his consequence-modeling code contains a data bank of inhalation factors that are calculated on the basis of assumed inhalation classes for radionuclides, classes that are thought to be characteristic of releases from LWRs.

9.4.2.7 Moisture

The present state of the art does not permit the realistic modeling of the behavior of wet plumes like those that might be emitted from an LWR during a severe accident. The reason given for this judgment is that the present understanding of radionuclide source terms does not include the ability to predict the state of any moisture that may be carried away from the reactor by a buoyant plume, whether entirely as vapor or as a fine aerosol. The greatest potential impacts on consequence modeling may well be due to a possible decrease in the predicted final height to which a hot plume rises. This decrease would be attributable to the loss of buoyancy after the evaporation of liquid droplets. Alternatively, the rainout of radionuclides after the condensation of water vapor on particles within the plume could result in the deposition of most of the radioactive material within a short distance of the reactor.

Current or planned research into radionuclide source terms will certainly include efforts to improve predictions of the spatial and temporal variations in the state of the water in the reactor system, and this should lead to an improved understanding of the quantity and the state of the water emerging into the atmosphere.

9.4.2.8 Release Categories and Their Frequencies

In the course of a PRA, it becomes necessary to choose a limited number of source terms for analysis. It is not unusual for a PRA to identify hundreds of accident sequences, and it is out of the question to examine all of these with codes like MARCH, CORRAL, and CRAC2. It is therefore essential to sort the sequences into a small number of groups for which the source terms are expected to be similar.

This grouping of accident sequences is a somewhat subjective exercise and is more properly the province of Chapters 7 and 8. However, the following guidelines are appropriate at this point:

1. Source terms of similar magnitude but with very different times of release or warning times should not be grouped together, as the effect of the evacuation procedure on the CCDF for early fatalities or injuries will be very different.

2. It is often possible to use arguments based on the relative probability of accident sequences to show that the public-risk contribution of one is necessarily much smaller than the contribution of the other.

The frequency of occurrence that is predicted for each radionuclide release category is the sum of the frequencies of the accident sequences in that category. The frequencies of the individual accident sequences are obtained from the quantification of event trees and from the subsequent containment analyses.

9.4.3 METEOROLOGICAL DATA

A code like CRAC2 requires a file of hourly meteorological data: wind speed, wind direction, stability category, and intensity of precipitation. The most obvious place to look for these data is at the reactor site, since each reactor site in the United States has a program of meteorological measurements, as required by the NRC's Regulatory Guide 1.23. For example, at one U.S. site the program of measurements includes (1) wind speed at 300 and 30 feet; (2) wind direction at 300 and 30 feet; (3) ambient temperature and dewpoint temperature at 30 feet; (4) temperature difference between the 300- and 30-foot and the 100- and 30-foot levels, measured with matched sensors; and (5) precipitation intensity.

After these data have been obtained, they must be processed into a form compatible with the consequence-modeling code that is being used. As mentioned above, CRAC2 requires a data file containing hourly values of wind direction, wind speed, stability category, and precipitation intensity. These hourly values are required for at least a year; ideally, more than one year of data should be used to increase the likelihood that all possible weather conditions are covered. Usually, the wind speed would be chosen for a height of 30 feet. As discussed in Appendix D1, a scheme based on two parameters, such as temperature difference and wind speed, should be used to determine the stability category, if possible. Finally, the precipitation intensity can obviously be determined from the precipitation measurements.

It is usually necessary to take the advice of a trained meteorologist while compiling the meteorological data file. Sometimes there are sequences of hours or even days when data are incomplete or inconsistent and expert judgment is required to fill in the gaps. Sometimes the site data are so bad that alternative sources must be found. Here again, a trained meteorologist should be consulted, and he will probably suggest the use of data from a nearby airport or from the National Weather Service. In looking for substitute meteorological data, it is clearly important to ensure that the meteorological characteristics of the substitute site are as close as possible to those of the reactor site. If there are terrain features that greatly affect the wind rose or considerable differences in precipitation patterns, the substitute may not be acceptable. If care is taken in choosing such data, however, experience in the use of codes like CRAC2 suggests that the errors introduced into the results of the consequence modeling are not great.

Other codes may require data in a different form. For example, CRACIT uses the wind speed and direction at more than one height. It can also use meteorological data from a number of sites in the neighborhood of a reactor. Each code manual should contain instructions detailing the data that are needed.

The user new to consequence modeling should note that the processing of the meteorological data looks deceptively simple. Problems inevitably arise, however, usually in connection with the accessibility, quality, and completeness of the data.

9.4.4 POPULATION DATA

Codes like CRAC invariably require that the population be assigned to the elements of a grid like that shown in Figure 9-8. Needless to say, the population grid should be taken consistent with the available meteorological data. Particular care should be taken to ensure that, when the meteorological data show the wind blowing toward, say, the north, it is the population north of the site and not south of the site that is assumed to be affected. This is a trap into which users of CRAC and CRAC2 often fall.

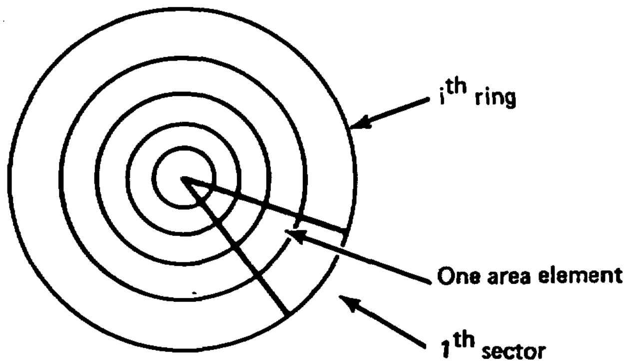

Figure 9-8. Population grid in CRAC.

The available data base is the U.S. census, 1970 being that which is currently available. (The 1980 data should be released soon.) These data give the number of people residing within a census enumeration district (CED). Where the CED lies entirely within an area element, it is simple to assign all of the population to that element. Where two or more elements lie within a CED, there are generally two alternative procedures. One is to assume that the population is uniformly spread over the CED. Another is to assign the population to some "center of gravity" of the CED. Each of these methods can assign people to places where there are none or underestimate the number of people in a given area. The problem becomes severe within 5 to 10 miles, where CEDs may overlap many grid elements. Within this region, it is advisable to carry out house counts and make use of aerial photographs.

Often, the data are required for years ahead, the predicted midlife of the plant being a typical example. This requires extrapolation, using known growth rates. These growth rates are usually available at the county level.

9.4.4.1 Transient Populations

In many areas, the population varies considerably with the time of year. Holiday resorts are an obvious example. In principle, this variation can be accommodated by collecting seasonal population distributions and causing the computer code to call the appropriate one for each weather sequence.

9.4.4.2 Diurnal Variations

There are large daily variations in population as people go to and return from work. The authors are not aware of any risk assessment in which this effect has been taken into account. In principle, information could be collected on places of work, schools, and the like. If a weather sequence (called by the CRAC2 sampling routine, for example) falls during working hours, the appropriate population distribution could be used. This variation in population could also affect evacuation and sheltering strategies. For example, people at work are more likely to be in or near large buildings with good sheltering characteristics than they are when at home (see Appendix E).

9.4.4.3 Computational Grid

In CRAC2, the population grid defines the computational grid since, within each element of the grid, such quantities as the plume-centerline airborne concentration are taken to be equal to the value calculated at the center of the grid. Other codes, such as TIRION (Kaiser, 1976), have computational grids that are not identical with the population grid. In general, consequence-modeling codes all introduce some form of computational grid within the framework of which certain approximations are made. An example is an approximation of the deposition calculation that is discussed in Section D3.2.1. In general, care should be taken to ensure that the calculational grid is not too coarse. If there is any doubt, calculations should be repeated with a finer grid. An instructive discussion of grids appears in the description of CRACIT, which uses a finer grid for dose calculations (Commonwealth Edison Company, 1981).

9.4.5 DEPOSITION DATA

The user of consequence models will probably find that his chosen code has available, in a data bank or in a standard input set, values of the dry-deposition velocity v_d and of the washout coefficient Λ for each nuclide.

As explained in Appendix D3, there are great uncertainties in the choice of dry-deposition velocities. The chosen code is very likely to assign v_d values of 1 or 0.3 cm/sec for particulate matter, and there is no reason for changing these values unless a sensitivity study is envisaged. In that case, v_d could be varied between 0.1 and 10 cm/sec. At the higher values of v_d, however, the models in some codes may become invalid.

In CRAC, Λ is assigned a value of 10^{-4} sec^{-1} for rainfall in neutral and stable weather conditions and 10^{-3} sec^{-1} in convective conditions. Sensitivity studies should allow Λ to vary from 10^{-5} to 10^{-2} sec^{-1}. If the user of consequence models wishes to justify different values of Λ, he will need a good understanding of what determines the value of Λ (e.g., particle size and rainfall characteristics); he will need to undertake a fairly ambitious review of work like that of Slinn (1977, 1978); and he will need to "benchmark" the changes to his code.

9.4.6 EVACUATION AND SHELTERING DATA

As has been explained, it is in the specification of input parameters for the evacuation model that the user can most influence the outcome of a consequence calculation. Accordingly, Appendix E is devoted to a thorough discussion of evacuation models, including the required input. The omission of a detailed discussion of evacuation and sheltering at this point is a deliberate decision on the part of the authors, who have provided Appendix E as a thorough, self-contained discussion of the subject.

9.4.7 ECONOMIC DATA

Perhaps the most useful thing that can be done here is to list as an example the data that CRAC2 requires as input.

9.4.7.1 Evacuation Cost

The modeler needs to provide the cost of evacuating a person and providing food and shelter for a few days, say, a week. These data can be obtained from such sources as an EPA study of 64 evacuations after disasters in the United States (Hans and Sell, 1974). The figure used in the RSS is \$95 per person; this would have to be updated to allow for inflation, as is true for all economic costs. According to the 1980 issue of the Statistical Abstract of the United States (published by the U.S. Department of Commerce), the Consumer Price Index rose from about 150 to about 225 between 1974 and 1980. An approximate 1980 estimate of the evacuation cost would thus be about \$158 per person. The contribution of evacuation costs to the total cost is simply the number of people evacuated times the individual evacuation cost.

9.4.7.2 Relocation Cost

The relocation cost essentially consists of an allowance for loss of income while an individual moves and finds a new job or while the corporation for which he works moves. It also includes a per capita allowance for household and business moving expenses. Chapter 12 of Appendix VI of the RSS (USNRC, 1975) gives details of how these components were shown to add up to \$2900 in 1975, and in CRAC2 the figure (updated to 1980) is \$4344. Again, CRAC2 simply multiplies this figure by the number of people who are relocated.

9.4.7.3 Value of Developed Property and Farm Property

In the RSS, the value of a property is assumed to be its market value. The CRAC2 model requires a per capita estimate of the value of depreciated residential, business, and public property over the interdiction period. The RSS figure for 1974 of $17,000 per head was obtained from the National Bureau of Economic Research; the authors of CRAC2 estimate $31,527 per head as a nationwide average. Again, the contribution to the total cost is the product of the number of people who are relocated and this per capita figure.

For farms, an estimate of the farm value per acre depreciated over the interdiction period is required. These figures vary from state to state and from county to county and can be found in such publications as Agricultural Statistics, published by the U.S. Department of Agriculture, for state averages and the County and City Data Book, published annually by the U.S. Department of Commerce, for county averages. Typical 1980 values for states vary from $100 per acre in New Mexico to $2222 per acre in New Jersey. Also required are the fractions of the area of each state that are devoted to farming. These can be found in the same Data Books.

In the RSS, the value of farmland was input state by state. In order to make use of this information, it is also necessary to identify the state to which each element of the grid of Figure 9-8 belongs. For site-specific studies, this may not be good enough. For example, the County and City Data Book for 1977 shows that, in the State of Arizona, the average value of farmland was $111 per acre. County by county, however, the values varied from $41 to $877 per acre.

If the reactor in question is situated in one of the wealthiest or poorest counties of its state, using the average for the state could give misleading answers. Hence, it may be necessary to identify elements of the population grid by county. Even within a county, the value of farmland may vary greatly; for example, farmland in the valley of a river flowing through a desert would be at a premium. Breaking the farm values down into sub-county areas requires the expenditure of considerable extra resources in the collection of local data, however.

9.4.7.4 Depreciation

If a property has an initial value made up of V_L, the value of the land, and V_I, the value of improvements, these improvements will deteriorate through the lack of maintenance at an annual rate of depreciation d_p. Assuming that the land itself retains its value in real terms, the value of the property after t_y years will be V_T, where

$$V_T = V_L + V_I \exp(-t_y d_p) \qquad (9\text{-}30)$$

CRAC2 requires input for d_p, which is judged to be about 20 percent for interdicted land (USNRC, 1975, Appendix VI, Section 12.4.2.1). For residential, business, and public property, V_I is usually valued at about 70 percent of $V_L + V_I$. For farm property, the corresponding figure is 25 percent. Essentially, CRAC2 calculates the difference between V_{T_I} and

$V_L + V_I$, where T_I is the interdiction period, and uses this as the basis for estimating the cost of interdicting land.

9.4.7.5 Crop Loss

As explained in Section 9.3.3.3, the CRAC2 code considers the ingestion of milk and crops, and works on the assumption that these will be destroyed if their consumption would deliver unacceptable radiation doses to various organs. For a given element on the CRAC2 spatial mesh, the annual sales value of farm produce per acre is required, and from this it is simple to calculate the total cost of destroying the crops. It is also necessary to know the fraction of farmland devoted to dairy products; from this and the average value of sales per acre the cost of destroying milk alone can be estimated. Additional input that is required consists of the seeding month and the harvesting month. If the deposition of radionuclides occurs outside the growing season, it is assumed that the crops are not to be destroyed. All this information is fed into CRAC2 state by state or county by county.

9.4.7.6 Fraction of Habitable Land

It is necessary to know the fraction of habitable land for each element of the population grid. It is essentially the fraction of the area of that element that is fit for human habitation and excludes mountains, lakes, rivers, and oceans. It is sufficient to estimate these figures crudely from a map.

9.4.7.7 Decontamination Costs

CRAC2 requires as input decontamination costs for farmland and developed land. The calculation of these costs is discussed in the RSS (USNRC, 1975, Appendix VI, Chapter 12).

The methods that would be considered for decontaminating farmland are to scrape the surface and dispose of it, to bury surface soil in place by grading, or to bury surface soil in place by deep plowing. The RSS estimates of costs for these activities were based on data from the Robert Snow Means Company (1974), Mohon (1974), and the U.S. Department of Agriculture (1974). A typical cost in the CRAC2 manual (Ritchie et al., 1981a) is $499 per acre (1980 dollars).

The methods that would be considered for the decontamination of developed property range from the firehosing of roofs and paving, and the replacement of lawns (which would give a decontamination factor of about 2) up to the replacement of roofs and paving (for a decontamination factor of about 20). A 1979 estimate of the cost of achieving a decontamination factor of 20 is $3349 per acre, taken from the CRAC2 manual.

9.4.7.8 Discussion

The foregoing inputs for economic costs serve to illustrate the kind of data that may be required if the chosen consequence-modeling code contains an economic impact routine. Table 9-12 summarizes some of the important inputs to the economic subgroup of a consequence model. These figures are examples only and are not necessarily valid for all applications of CRAC2.

Table 9-12. Examples of important input to the economic subgroup of CRAC2[a,b]

Evacuation cost per person	$158
Relocation cost per person	$4344
Value of developed property, per person	$31,527
Decontamination cost for developed property (DF 20), per person	$3349
Decontamination cost for farmland (DF 20), per acre	$499
Depreciation rate per year for developed property (fraction of value)	0.2
Value of farm property (state averages), per acre	From $100 (New Mexico) to $2222 (New Jersey)
Value of annual farm sales (state averages), per acre	From $15 (Wyoming) to $500 (Delaware)
Fraction of sales--dairy products (state averages)	From 0.024 (Wyoming) to 0.791 (Vermont)
Fraction of land devoted to farming (state averages)	From 0.077 (Maine) to 0.795 (Illinois)

[a]From the CRAC2 user's manual (Ritchie et al., 1981a).
[b]All figures are in 1980 dollars.

9.4.8 HEALTH PHYSICS

The health-physics calculations carried out by a code like CRAC2 require vast quantities of information (e.g., the inhalation-dose-conversion factors described in Section 9.3.3.1). A typical run of CRAC2 will make use of up to 54 radionuclides and 13 body organs, with the inhalation factors calculated for seven time periods--a total of some 5000 numbers. Most consequence-modeling codes have a data bank containing these quantities. The drawback to this is that the modeler has then imposed on the user his own views about the chemical form, the particle-size distribution, and other properties of the released radionuclides.

It is instructive to review the information required in a code like CRAC2, should the user wish to input his own information on various parameters associated with the health-physics calculations.

9.4.8.1 Inhalation Factors

Among the data files in CRAC2 is one containing dose-conversion factors for the inhalation pathway, $F^{i}_{n,k}(t)$. Here (n,i) identifies the radionuclide in question (i.e., the i^{th} daughter of the n^{th} radionuclide-decay chain). As already mentioned, there are 54 such nuclides. The subscript k identifies the organ, of which there are 13: lung, bone marrow, skeletal bone, endosteal cells, stomach wall, small intestine, upper large intestine and lower large intestine, thyroid, whole body, testes, ovaries, and "other" tissues.

The variable t identifies the time periods, of which there are seven: (1) period for acute exposure (1 year for the lung; 7 days for the marrow, skeletal bone, endosteal cells, stomach wall, small intestine, upper large intestine and lower large intestine; 2 days for the thyroid, whole body, testes, ovaries, and other tissues); (2) 1 year; (3) 1 to 10 years; (4) 10 to 20 years; (5) 20 to 30 years; (6) 30 to 40 years; and (7) 40 to 50 years.

All of the above quantities are required in various dose-response relationships. (See Section 9.4.2.5 for a discussion of inhalation factors and particle sizes.)

9.4.8.2 Dose-Conversion Factors: External Irradiation

CRAC2 contains the quantity $G^{i}_{n,k}(t)$, which is an array containing the dose-conversion factors (rem/Ci-m^2) for exposure to contaminated ground. As before, (n,i) identifies the radionuclide and k the organ; t is a variable specifying (1) the 8-hour integrated dose delivered to organ k by isotope (n,i); and (2) the 7-day integrated dose. Also contained in this array are dose-rate-conversion factors (rem·m^2/Ci·yr). These factors are used for calculating chronic groundshine doses. The doses are obtained by multiplying the initial deposited activity of each radionuclide by the corresponding element of $G^{i}_{n,k}(t)$.*

CRAC2 also contains a quantity, $C^{i}_{n,k}$, giving the radiation dose accumulated by organ k as a result of exposure to 1 Ci-sec/m^3 of radionuclide (n,i), that is, the dose-conversion factor for cloudshine.

Note that the elements of $G^{i}_{n,k}(t)$ are calculated with an infinite-plane approximation and the elements of $C^{i}_{n,k}$ are calculated with a semiinfinite-cloud approximation. As explained in Section 9.3.3.2, cloud-shape correction factors must subsequently be applied.

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*Care must be taken to treat daughter buildup correctly. If, as stated above, the doses are obtained by multiplying the initial deposited activities by $G^{i}_{n,k}(t)$, then radioactive decay is not explicitly calculated. The quantity $G^{i}_{n,k}(t)$ should implicitly take account of daughter buildup over the time t. If this is not done, some radiation doses can be significantly underestimated.

If the code in question makes direct use of an integral over the cloud or ground to calculate the above quantities, information must be supplied on the exposure-buildup factor and on the gamma-ray spectrum of each radionuclide. Kaiser (1976) shows how this can be done.

9.4.8.3 Computation of Early Health Effects

Figure 9-9 illustrates the information required for each early health effect that is considered. The quantity D_{jk} contains four dose-limit values (J = 1-4), with k specifying the organ in question. The quantity D_{1k} is the threshold below which the probability of the effect is zero; D_{4k} is the dose value above which the probability of the effect is 1; D_{2k} and D_{3k} are intermediate values corresponding to the probabilities given by P_{1k} and P_{2k}, respectively. The quantities D_{jk} and P_{jk} thus specify the probability of the effect over the entire dose range. The model assumes that the points described by D_{jk} and P_{jk} are connected by straight lines. The quantity P_{jk} contains two probabilities at which the slope of the dose-response relationship changes.

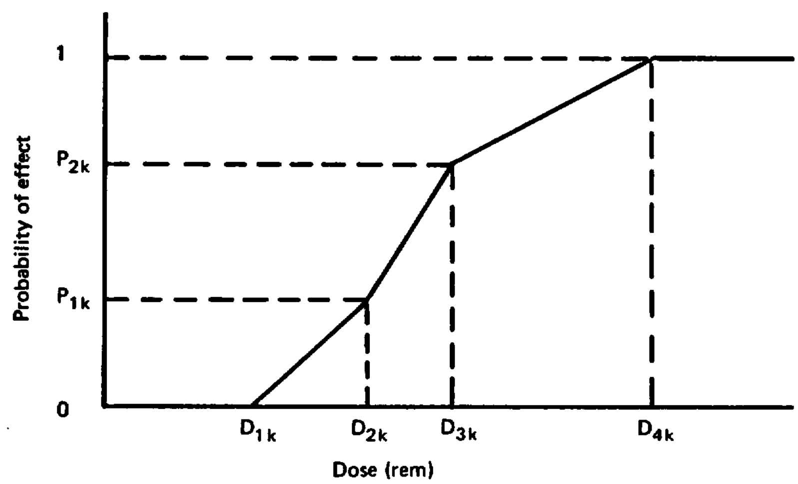


Figure 9-9. Dose-response model.

To give examples, the greatest cause of early fatalities is generally the accumulation of radiation dose in the bone marrow, where the radiation dose is specified by 0.5 (7-day + 30-day doses from inhalation) + dose delivered by the passing cloud + dose delivered within 7 days by deposited gamma emitters. The dose-limit values are 320, 400, 510, and 615 rem. The two probabilities contained in P_{jk} are .03 and .5. This is a simulation of the dose-response relationship with "supportive medical treatment" described in Appendix VI of the RSS.

For acute injuries, irradiation of the whole body is generally the most important indicator of whether hospitalization is required. The whole-body radiation dose, including the inhalation plus external-irradiation components, is used in a dose-risk relationship for which the dose limits are 55, 150, 280, and 370 rem. The two associated probability limits are .05 and 1.

Some modelers introduce more complicated dose-risk relationships with more than four dose limits or different shapes. Fryer and Kaiser (1979) give examples.

9.4.8.4 Computation of Latent Effects from Early Exposure

The key to the computation of latent-cancer effects is a population-dose conversion factor. The product of this factor and the radiation dose accumulated in a given organ gives the probability that cancer will develop in that organ. Some codes apply a single conversion factor to a single dose accumulated over a 50-year period. Typical values are 2×10^{-5} per rem delivered to the lung for lung-cancer induction and 10^{-5} per rem delivered to the bone marrow for leukemia (see Fryer and Kaiser, 1979). These figures, taken from the publications of the United Kingdom's National Radiological Protection Board, are consistent with the deliberations of the International Commission on Radiological Protection.

CRAC2 is somewhat more sophisticated in that it requires conversion factors for radiation doses delivered over different periods of time. An example for lung cancer is as follows:

1. 2.2×10^{-5} for radiation doses delivered during the first year.
2. 2.2×10^{-5} for years 1 to 10.
3. 2.2×10^{-5} for years 11 to 20.
4. 1.5×10^{-5} for years 21 to 30.
5. 8.1×10^{-6} for years 31 to 40.
6. 4.0×10^{-6} for years 41 to 50.
7. 1.5×10^{-6} for years 51 to 60.
8. 2.2×10^{-7} for years 61 to 70.

These figures reflect the fact that radiation doses delivered 50 to 60 years after the accident would manifest themselves as cancers, if at all, at least 2 to 40 years later, by which time the individual in question would almost certainly have died for other reasons; that is, these figures reflect the changing age distribution of the population that was initially exposed to the radioactive plume.

CRAC2 also allows a variation of the linear hypothesis, the central estimate discussed in Section 9.3.5. Typically, the dose-effectiveness factors would be reduced by a factor of 5 for radiation doses not exceeding 30 rem and by a factor of 2.5 for doses between 30 and 300 rem. As explained in Section 9.3.5, this is not yet a completely proved procedure. The reason it was used in the RSS is as follows: the risk estimates based on the linear extrapolation are taken from the BEIR I report (NAS-NRC, 1972). In 1975, however, the National Council on Radiation Protection and Measurements (1975) issued a report warning that the BEIR I estimates, which were derived from large radiation doses at high dose rates, are very likely to overestimate the risk from low radiation doses delivered at low dose rates. The use of the central estimate was intended as a more realistic estimate of risk, and as has been discussed in Section 9.3.5.2, the central estimate is consistent with the linear-quadratic model of the

BEIR III report (NAS-NRC, 1980), which is suitable for estimating the somatic effects of radiation with a low linear energy transfer.

9.4.8.5 Chronic Effects

As for the data needed for calculating chronic effects, perhaps the most helpful discussion for the potential user of consequence models is a brief summary of the requirements of CRAC2, which considers (1) inhalation of resuspended particles, (2) ingestion of exposed crops, (3) ingestion of milk products, (4) ingestion of milk, (5) ingestion of crops contaminated by root uptake, and (6) exposure to contaminated ground. For each of these pathways, CRAC2 requires a list of the nuclides considered important; for example, for milk ingestion these are generally taken to be I-131 and I-133. The radiation doses delivered by pathways 1 and 6 are then calculated by the methods described in Section 9.3.3.4 (resuspension) and Section 9.3.3.2 (radiation dose from deposited gamma emitters).

For the ingestion pathways, further information is required in the form of concentration factors (see Section 9.3.3.3) relating the activity ingested by a typical individual to the initial deposited activity, for each of the radionuclides identified as important for the ingestion pathway in question. Also required, for each of the nuclides being considered in all of the ingestion pathways and each of the 13 organs identified in Section 9.4.8.1, are the ingestion factors (see Section 9.3.3.3) for six time periods, measured from the time at which the ingestion took place: 0 to 10, 11 to 20, 21 to 30, 31 to 40, 41 to 50, and more than 50 years.

For each exposure pathway, the weathering half-life is required, if relevant; also required is the allowable limit of dose accumulation, together with the period of time over which the dose is accumulated. These limits are obtained from the publications of the U.S. Federal Research Council (FRC, 1964, 1965) and the British Medical Research Council (MRC, 1975), as explained in Section 9.3.3.3. Examples are as follows:

1. Inhalation of resuspended radionuclides--15 rem delivered to the lung over 70 years.

2. Ingestion of exposed crops--3.3 rem delivered to the whole body over 1 year.

3. Ingestion of milk products--3.3 rem delivered to the bone marrow over 1 year.

4. Ingestion of milk--10 rem delivered to the thyroid over 1 year.*

*In NUREG-0396 (Collins et al., 1978), in the section discussing the emergency-planning zone for ingestion, the size of the zone is based on an expected revision of milk-pathway Protective Action Guidelines by the Food and Drug Administration's Bureau of Radiological Health. The recommended guidelines were supposed to be as low as 1.5 rem to the infant thyroid, but, to the authors' knowledge, these guidelines have never been officially published.

5. Ingestion of crops contaminated by root uptake--5 rem delivered to the bone marrow over 10 years.

6. Exposure to deposited gamma emitters--25 rem delivered to the whole body over 30 years.

In general, for accident sequences in LWR plants, it is predicted that milk ingestion and external exposure are the most important of the chronic exposure pathways, at least in the United States. However, chronic exposure may be highly dependent on agricultural practices and the patterns of food consumption, which may change the relative importance of certain radionuclides. If a probabilistic risk assessment is being carried out for a reactor site in another country, different pathways may need to be considered. In countries with a predominantly Chinese population, for example, the ingestion of milk is negligible, and it is conceivable that other pathways of chronic exposure, such as the contamination of fish farms or duck farms, would be important.

9.4.8.6 Discussion

It is apparent that the user of a consequence-modeling code like CRAC2 has a time-consuming task on his hands if he wishes to input new health-physics data. In general, the health physics of the Reactor Safety Study is still considered to be acceptable by the consequence-modeling community. However, the Nuclear Regulatory Commission is at present considering the possibility of setting up a health-physics data bank that would be freely accessible to all who wish to use it. Such a data bank is needed because the typical user of consequence models will have neither the time nor the expertise to make significant changes to the health-physics data contained in a code like CRAC. Meanwhile, a fresh compilation of inhalation factors (the same goes for ingestion factors and concentration factors) would require a fairly thorough survey of the available literature on health physics (see Sections 9.3.3 and 9.3.5).

9.4.9 DISCUSSION OF DATA REQUIREMENTS

The above review of inputs to consequence models is not comprehensive since different codes have different data requirements that cannot all be discussed here. The potential user should realize by now, however, that he is faced with a considerable amount of work and must devote thought to a variety of topics before he can begin to run his code. The amount of data required is so great, and the purposes for which it is needed cover such a range of topics, that it would be foolish to try to use the code as a "black box." For a sensible preparation of the input, a good background knowledge is required.

9.5 PROCEDURES AND FINAL RESULTS

9.5.1 PROCEDURES

The procedures presented here are aimed at a user of consequence models who has his code up and satisfactorily running on a computer.

9.5.1.1 Deciding on the Purpose of the Consequence Analysis

This involves selecting from among the list of applications given in Section 9.1.2 or devising another application not mentioned there. By doing this, the user will determine (1) which of the input options available in his code he needs to use, (2) which data sets he needs to collect, and (3) which output options he needs to exercise.

9.5.1.2 Collection of Data

As has been explained, this is a major undertaking and involves data collection in many areas.

Basic Radionuclide Data. A reactor-core inventory should be compiled for a selected list of radionuclides, together with a list of associated data, as described in Section 9.4.1.

Source-Term Data. The analyst should consult with the workers on other tasks (quantification, physical processes of core-melt accidents, radionuclide release and transport) in order to compile a table of the properties of radionuclide source terms, such as appears in Table 9-1 and is discussed in Section 9.4.2.

Meteorological Data. A tape containing hourly meteorological data for one or preferably more years should be obtained from the reactor site. In consultation with a meteorologist, the code user should decide whether the quality of the data is adequate. If not, substitute data from a nearby site should be obtained. The data set should then be processed into the form required for the computer code (see Section 9.4.3). If meteorological data from multiple stations are being used, the exercise should be repeated for each meteorological station.

Population Data. After the year for which population data are needed (e.g., plant midlife) and the radii of the elements of the population grid (see Figure 9-8) are selected, data from the U.S. census and/or the FSAR should be processed to assign the population to these elements. For some U.S. reactors, it may be necessary to obtain data from Canada or Mexico. Experience indicates that the processing of population data can be one of the more time-consuming and costly elements of a consequence analysis.

Deposition Data. Dry-deposition velocities and washout coefficients should be assigned to each radionuclide (see Section 9.4.5).

Evacuation and Sheltering Strategies. The choice of data for the evacuation model is comprehensively discussed in Appendix E. Table E-3 summarizes the kind of input that is required. In order to perform this task, site emergency plans should be consulted, together with any associated studies that give estimates of such quantities as evacuation time and effective evacuation speed, sheltering factors, and ventilation.

Economic Data (usually optional). The collection of economic data is discussed in Section 9.4.7, and examples of important input parameters are given in Table 9-12. It is important to note that it may be necessary to collect data on farming input variables at the county or smaller level. The statewide averages of the value of farmland per acre given in the CRAC2 user's manual, for example, may differ greatly from county averages.

Health-Physics Data. As indicated in Section 9.4.8, these data are usually available in a data bank associated with the code or in a standard input set. It is unlikely that a user would wish to undertake the laborious task of replacing the data bank with his own figures.

9.5.1.3 Exercising the Code

In principle, exercising the code should be straightforward. The bulk of the work should have been done in the collection and the processing of data. The code should be run for the base case and for other cases that may have been devised to test the sensitivity of the results to variations in input data.

9.5.1.4 Interpreting the Output and Writing the Report

The output is fully described in Section 9.5.2. Advice on report writing is given in Section 9.7.

9.5.2 FINAL RESULTS

The final results of a consequence analysis are also the final results of the complete probabilistic risk assessment. Hence, the results should be presented in probabilistic form, and this is why the characteristic output of a consequence analysis is in the form of a CCDF, examples of which appear in Figure 9-1.

The CCDF is a compilation of the results of many separate calculations. A code like CRAC2 essentially repeats the same calculation many times, changing each of the following variables over its full range of values:

1. The accident sequence or category (e.g., PWR categories 1 through 9 in Table 9-1).

2. The weather sequence (each set of hourly values of stability category, wind speed, and precipitation intensity).

3. The wind direction.

For each category, weather sequence, and wind direction, a code like CRAC2 will calculate the predicted number of early fatalities in the sector toward which the wind is blowing (and, of course, early injuries, latent-cancer fatalities, etc.). With this predicted number can be associated a frequency, which is the product of the frequency of occurrence of the accident sequence or category and the joint probability of occurrence of the weather sequence and the wind direction.

In principle, the calculation can be broken down still further. For example, CRAC2 allows the user to implement up to six evacuation schemes, each with an associated probability. CRACIT allows an effectively unlimited number of evacuation schemes. It is possible to envision calculations in which the deposition velocity is a variable with an associated probability distribution (Beyea, 1978a,b), or in which there could be a choice of dose-response relationships, each with an associated probability.

Such calculations yield pairs of numbers (e.g., the predicted number of early fatalities given an accident category, weather sequence, wind direction, evacuation scheme) together with the frequency with which that combination of variables is predicted to occur. Taking all such pairs of numbers for all possible combinations of the variables gives a frequency distribution that can be readily cumulated to give a CCDF.

As can be seen from Figure 9-1, CCDFs for early fatalities and latent-cancer fatalities are among the possible output of a consequence analysis; indeed, these are among the most frequently used. Other possible CCDFs include the following, with examples taken from the RSS:

1. Early illness, which is essentially defined by reference to a whole-body dose large enough to cause hospitalization (Figure 9-10).

2. Genetic effects (Figure 9-11).

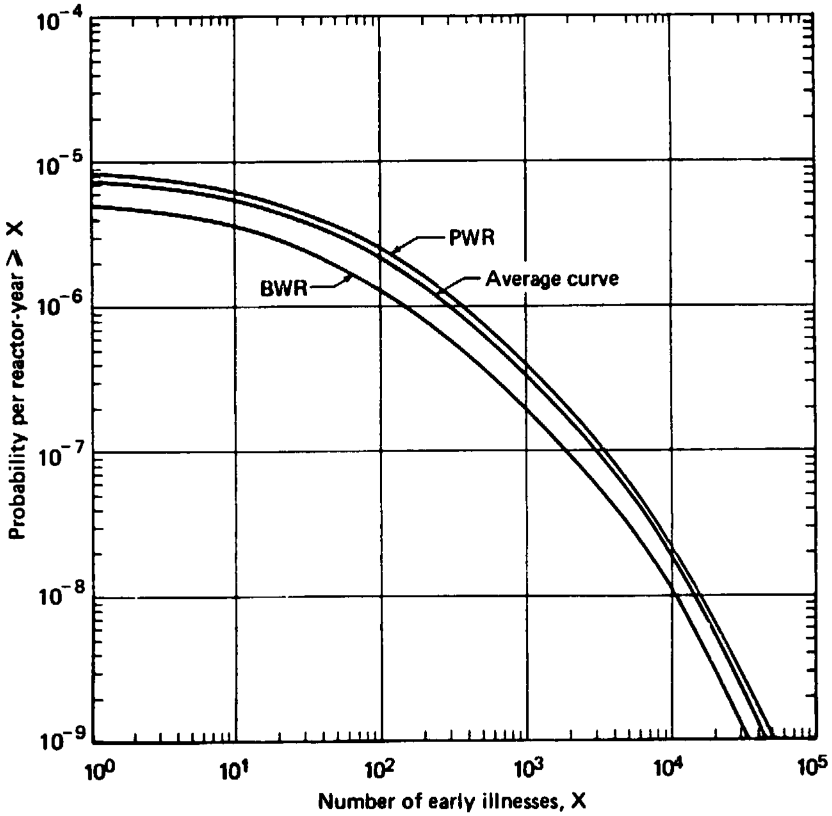
3. Areas requiring decontamination or relocation (Figure 9-12).

4. Property damage (Figure 9-13).

The CRAC2 user's manual lists a great variety of possible outputs from a consequence analysis, all of which can be expressed in the form of CCDFs. Examples are as follows:

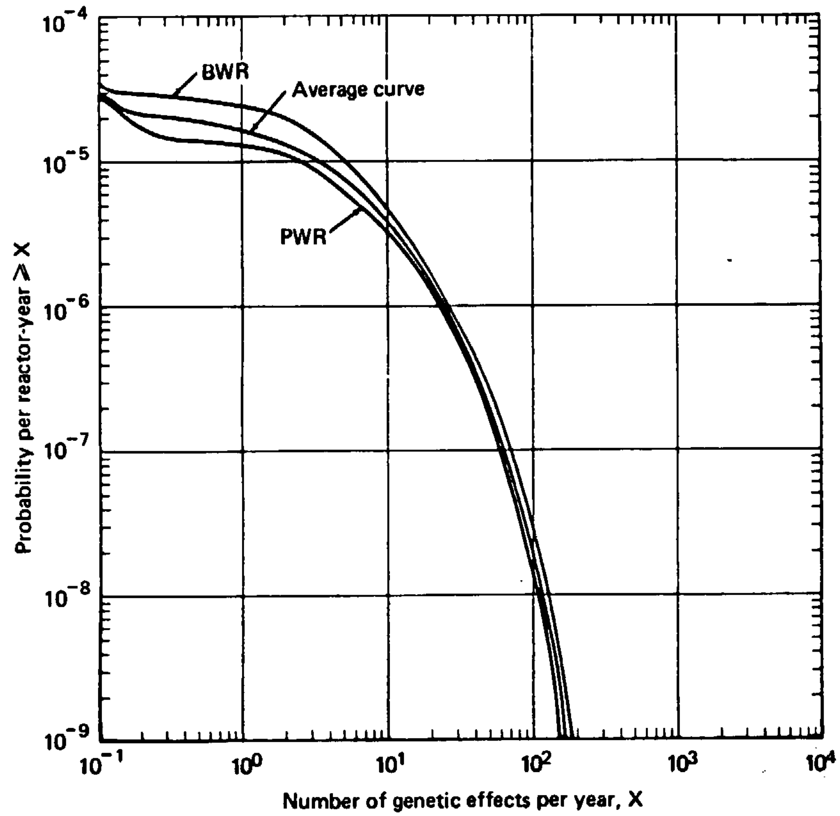
1. Number of people with an acute bone-marrow dose exceeding 200 rem (this number is of interest because it indicates how many people would require hospital treatment).

2. Risk of early fatality at the midpoint of each of several specified radial intervals on the computational or population grid.



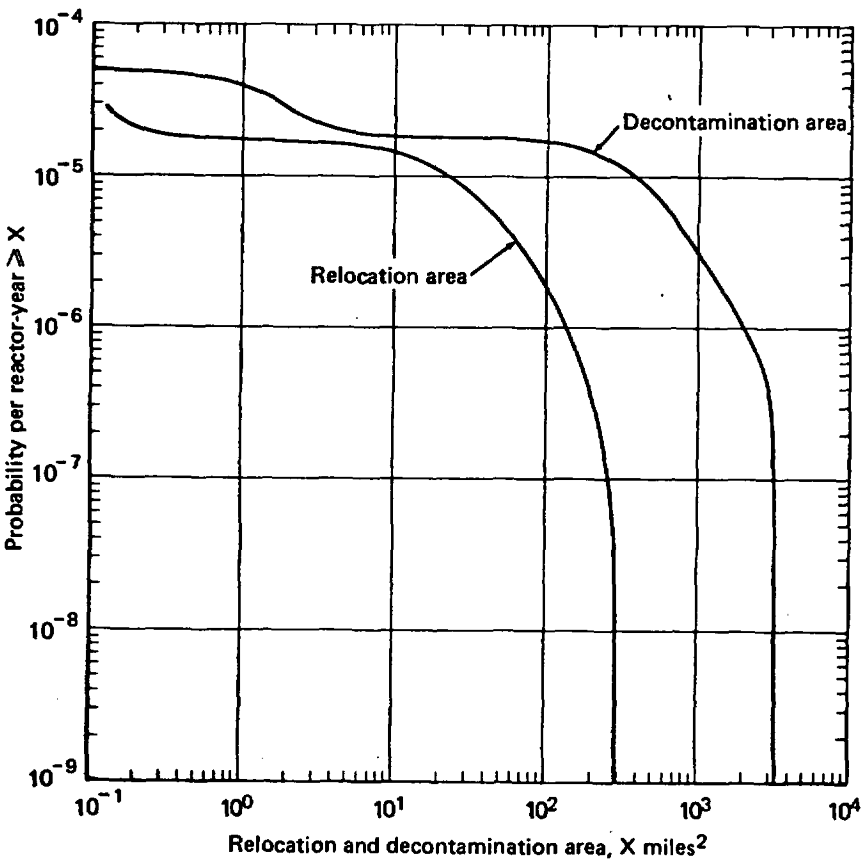
Note: Approximate uncertainties are estimated to be represented by factors of 1/4 and 4 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

Figure 9-10. Complementary cumulative distribution function for early illnesses. From the Reactor Safety Study (USNRC, 1975).



Note: Approximate uncertainties are estimated to be represented by factors of 1/3 and 6 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

Figure 9-11. Complementary cumulative distribution function for genetic effects per year. From the Reactor Safety Study (USNRC, 1975).



Note: Approximate uncertainties are estimated to be represented by factors of 1/5 and 2 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

Figure 9-12. Complementary cumulative distribution function for relocation and decontamination area. From the Reactor Safety Study (USNRC, 1975).

3. Greatest distance from the reactor at which acute fatalities or injuries are predicted to occur.

4. Number of people residing in the area that would need to be permanently interdicted (for more than 30 years).

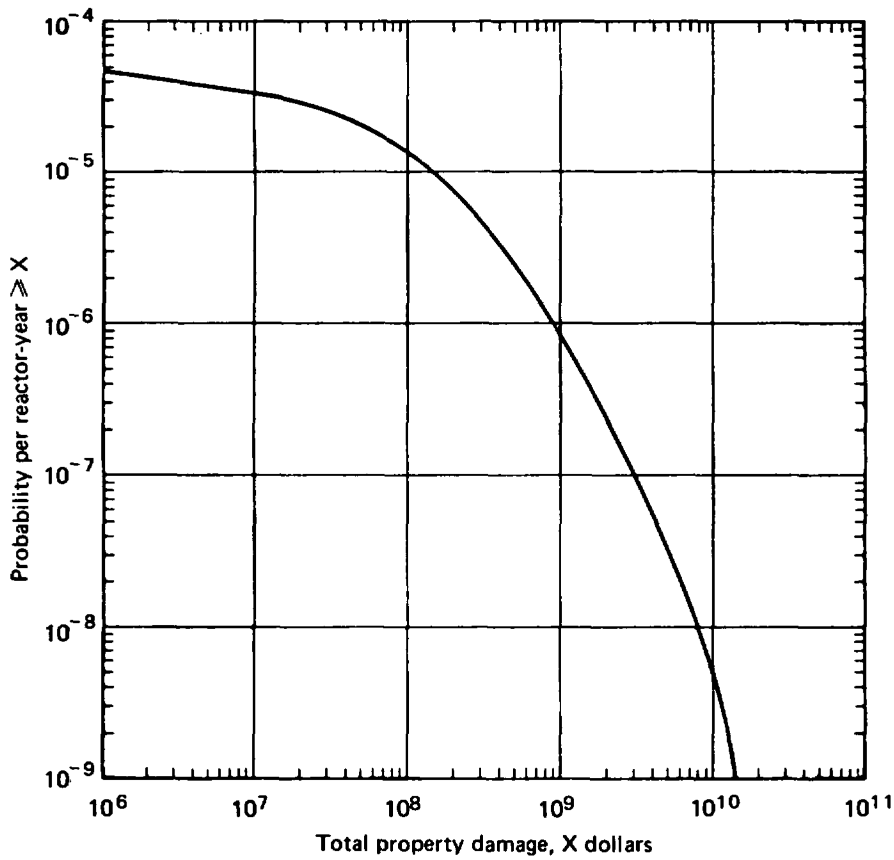
5. Cost of permanent land interdiction.

6. Total land area permanently interdicted.

7. Cost of contaminated-milk disposal.

In all, CRAC2 allows the user to choose 84 output options for display as CCDFs; the above list is a sample of the possibilities.

Consequence models are also able to output the individual risk of early fatality or latent-cancer fatality as a function of distance. Examples are



Note: Approximate uncertainties are estimated to be represented by factors of 1/5 and 2 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

Figure 9-13. Complementary cumulative distribution function for total property damage. From the Reactor Safety Study (USNRC, 1975).

given in Figure 9-14. These individual risks have recently received renewed attention because the NRC has published draft safety goals that contain target values for individual risk, an example being 5 x 10^{-7} per year for early fatality, averaged out to 1 mile (USNRC, 1982a). Since the implementation of these safety goals had not been finalized at the time of writing, however, it is premature to describe how individual risks should be calculated in the context of safety goals. It is important to be aware that there are many uncertainties; for example, the individual risk of early fatality is extremely sensitive to assumptions about evacuation and plume rise.

Consequence models can also be used to obtain output for given weather conditions, such as the radiation dose received by a given organ as a function of position. The user who so desires may, if he uses all the options available in his user's manual, produce an overwhelming stack of computer output. It is desirable to exercise a certain degree of restraint in the choice of a sensible number of output options.

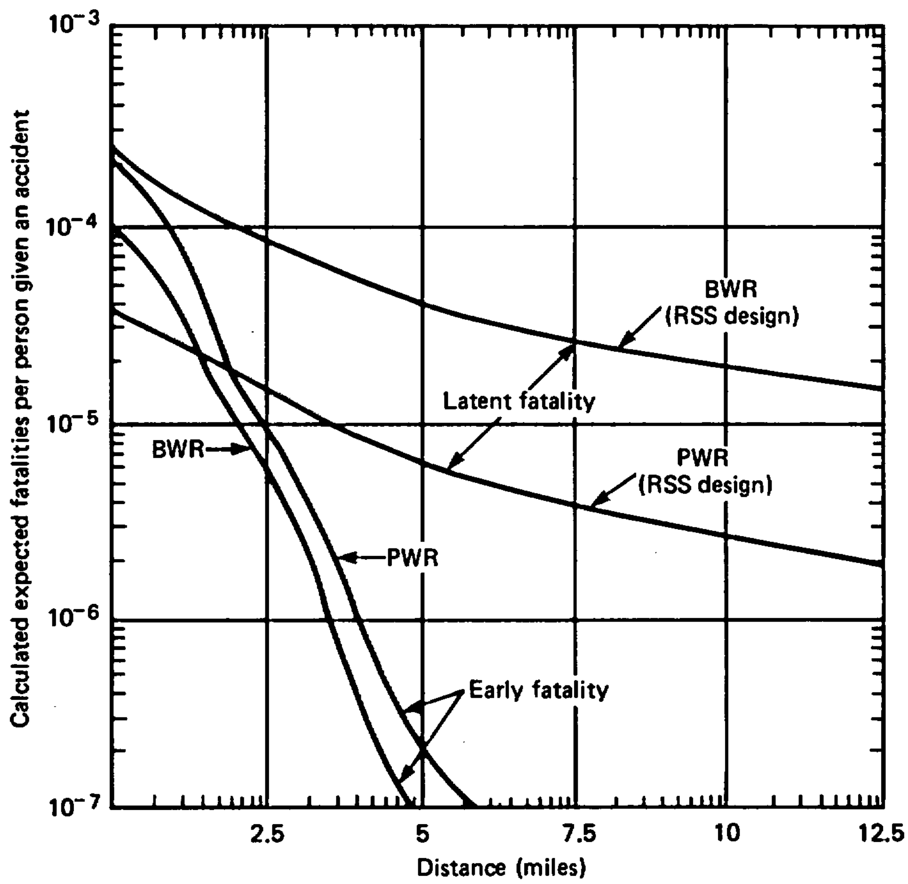


Figure 9-14. Calculated risk to an individual of early and latent fatality as a function of distance from the reactor for the accidents described in the Reactor Safety Study (including non-core-melt accidents). The latent fatalities are those attributed to immediate exposures; no chronic effects from long-term groundshine or long-term pathways are included. From NUREG-0739 (USNRC, 1980).

9.6 ASSUMPTIONS, SENSITIVITIES, AND UNCERTAINTIES

A necessary element in the interpretation of the results of a consequence analysis is an estimate of the uncertainties associated with the results.

More recent studies of uncertainties in PRA tend to produce CCDFs that look like that shown on Figure 9-15. Ideally, the upper and lower bounds are expressed in statistical terms as confidence limits, perhaps at the 5- and 95-percent levels. The median is the curve above or below which the true CCDF is equally likely to lie. The mean is shown as lying close to the 95-percent limit. This is characteristic of the highly skewed probability distributions (e.g, the lognormal) that are currently being derived to express uncertainties in PRAs. As noted by the Risk Assessment Review Group

(Lewis et al., 1978), uncertainties are larger than those given in the RSS. Typical ranges of uncertainty span two orders of magnitude or more (Philadelphia Electric Company, 1981; Commonwealth Edison Company, 1981).

It is convenient to divide the factors contributing to these uncertainties into two parts. The first consists of the factors deriving from other parts of the PRA exercise; these are discussed in the appropriate chapters and in Chapter 12. The second consists of those uncertainties that are peculiar to consequence analysis.

The parameters and modeling assumptions used at various stages in a consequence model are reviewed below in the context of their contribution to uncertainties in CCDFs. If a parameter or modeling assumption is said to be a major contributor to uncertainty, this means that, when the parameter or modeling assumption is varied over its plausible range, there is a broad band within which the corresponding CCDF may lie. A major contributor to uncertainty is one for which this band is as much as a factor of 10 in breadth (perhaps even more).

The decision as to whether a particular modeling assumption or parameter makes a major, moderate, or small contribution to uncertainty is the subjective judgment of the authors. Where possible, this judgment is based on sensitivity studies, that is, studies in which the modeling assumption or parameter value is changed and the CCDF recalculated in order to see how it varies. Hence, the uncertainties discussed in this section are not quantitative uncertainties in the sense that the bounds on CCDFs are expressed as confidence limits as shown in Figure 9-15, but are quantities that themselves have meaning in a subjective sense only.

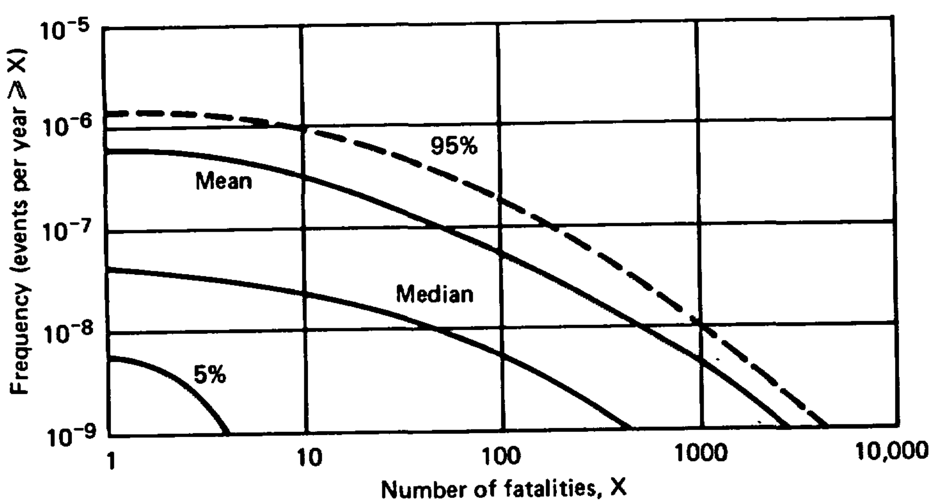


Figure 9-15. Typical uncertainty bounds on a CCDF for early fatalities.

9.6.1 INVENTORY OF RADIOACTIVE MATERIAL

The available methods for calculating the inventory of radioactive material in the reactor core at the time that the chain reaction ceases are well established. The ORIGEN code (Bell, 1973; Croff, 1980) is widely used and takes account of all the important processes--fission, radionuclide decay and daughter buildup, and neutron activation. The factors that have an important influence on the outcome of a consequence analysis are summarized in Table 9-13.

It is apparent that the user should be as realistic as possible in his choice of power level and burnup; he should also distinguish between PWRs and BWRs. Once these choices have been made, the residual uncertainties in the radioactive inventory are small. Hence the radionuclide inventory is a small contributor to the uncertainties in CCDFs.

9.6.2 SOURCE TERMS

The methods for calculating source terms are described in Chapters 7 and 8 of this guide, and a summary of important uncertainties appears in Table 9-14. It is in this area that many of the greatest uncertainties in consequence modeling arise, and it is worth devoting some attention to the factors designated as having a major influence on uncertainty.

9.6.2.1 Magnitude of the Source Term

Figure 9-16 shows an example of the effect predicted for the early-fatality CCDF if the source terms used in the RSS are reduced by factors of 5 or 10. Table 9-15 shows the impact of the same reductions on early injuries, latent-cancer fatalities, and areas interdicted for 10 years or more. Some authors have argued strongly that similar or even larger reductions in source terms are justifiable on the basis of existing evidence (Levenson and Rahn, 1981; Morewitz, 1981). Others argue that such reductions are not proved (USNRC, 1981b; Passadeg et al., 1981; Levine et al., 1982). All authors agree that there are large uncertainties in the magnitude of the source term, however.

The authors of the Limerick PRA study (Philadelphia Electric Company, 1981) identified a lack of knowledge about the mechanisms of source-term attenuation within the reactor-coolant system and the containment as the most important uncertainty in predicting the magnitude of the consequences. The authors of the Zion study (Commonwealth Edison Company, 1981) attempted to model the source-term uncertainty by assigning a probability distribution to the magnitude of release for each accident category. This was done by a process of subjective judgment, however. It is clear that the question of uncertainties in this important area remains to be settled and that considerable future research is required in order to quantify and reduce the uncertainties.

Table 9-13. Radionuclide inventory: sensitivities and uncertainties

Parameter or modeling assumption	Quantity most directly sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs			Sensitivity studies	Relevant section of this chapter
		Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage		
PWR vs. BWR (modeling)	Inventory of long-lived radionuclides	Low	Low	Low	Ostmeyer (1981)	9.4.1
Power level	Inventory of radio-nuclides	Low	Low	Low	Ostmeyer (1981)	9.4.1
Burnup	Inventory of long-lived radionuclides	Low	Low	Low	Ostmeyer (1981)	9.4.1

Table 9-14. Source terms: sensitivities and uncertainties

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs			Sensitivity studies	Relevant section of this report
		Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage		
Magnitude of source term (modeling and parameter)	Airborne and deposited levels of radioactive material	Major	Major	Major	RSS (USNRC, 1975) Wall et al. (1977) Zion study[a]	9.4.2.2, Chapter 7, Chapter 8
Frequency of occurrence of each sequence/category (modeling and parameter)	Frequency of CCDFs	Major	Major	Major	Limerick study[b] Zion study[a] Erdmann et al. (1981)	9.4.2.1
Time of release (parameter)	Time available for evacuation	Major (except peaks)	Low	Low		9.4.2.3
Duration of release (parameter)	Plume width Possibility of wind shift or weather change during release	Major (especially peaks)	Low	Low	Griffiths (1977) Benchmark exercise Zion study[a]	9.4.2.3
Warning time (parameter)	Time available for evacuation	Major (except peaks)	Low	Low		9.4.2.3
Building wake or dimensions of release (parameter)	Airborne concentration near reactor	Low	Low	Low		9.4.2.4
Rate of energy release (parameter and modeling)	Height of plume rise	Moderate to major for some sequences (low for peaks)	Low	Low	Russo (1976) Russo et al. (1977) Kaiser (1977, 1981)	9.3.1.5
Particle-size distribution (parameter and modeling)	Deposition velocity Washout coefficients Dose-conversion factors	Moderate	Moderate	Major	Kaul (1981b) Benchmark exercise Hunt et al. (1979)	Appendix D3, 9.4.2.6
Chemical form (parameter)	Dose-conversion factors Deposition velocity	Moderate	Moderate	Moderate	Hunt et al. (1979)	9.4.2.7

[a]Commonwealth Edison Company (1981).
[b]Philadelphia Electric Company (1981).

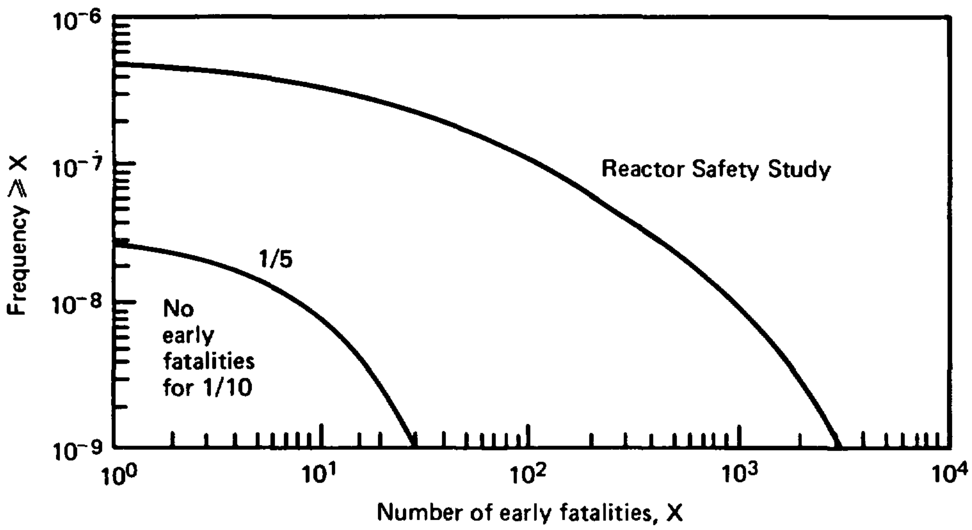


Figure 9-16. Perspective on the risk predicted by the Reactor Safety Study: iodine and particulate releases to the atmosphere reduced by factors of 5 and 10. Same modeling as was used in the Reactor Safety Study. Note that the reductions shown can be sensitive to details of modeling, such as evacuation. From C. Starr, M. Levenson, and I. B. Wall, "Realistic Estimates of the Consequences of Reactor Accidents," USNRC briefing, 1980.

Table 9-15. Impact of decreasing the magnitude of the release[a]

	Assumed reduction in quantity of iodine and particulates released to the atmosphere		
Consequence	RSS	1/5	1/10
Early injuries	1	0.032	0.0020[b]
Latent-cancer fatalities	1	0.35	0.22[c]
Area interdicted 10 years	1	0.11	0.037[c]

[a]From Starr et al. (1980). Reference for comparison is the Reactor Safety Study (USNRC, 1975).

[b]Nonlinear because of thresholds in early effects.

[c]Nonlinear because both depend on interdiction and decontamination measures, which are effectively threshold effects.

9.6.2.2 Frequency of Occurrence of Each Category

Uncertainties in the frequencies of occurrence predicted for accident categories or sequences propagate directly into uncertainties of comparable magnitude on the frequency axis of CCDFs. The source of these uncertainties is to be found in the chapters on the quantification of event trees and has nothing to do with consequence analysis per se; however, this uncertainty is the single greatest contributor to uncertainties on the frequency axis of CCDFs.

9.6.2.3 Duration of Release

In the context of uncertainty, the most important effect of an increased duration of release is to allow the possibility of a change in wind direction and/or weather conditions while the release of radioactivity is taking place. Figure D-12 and the accompanying text show the possible effect of wind shift on the dispersion of the plume. This behavior is to be contrasted with that of the single puff described in Figure D-10. A comparison of these two figures shows that the wind shift causes the plume of longer duration to spread out over a much wider area than that covered by the puff. In principle, this could cause the radiation dose delivered by the extended plume to fall below thresholds for early effects. That is, the long duration of release may introduce considerable conservative bias into the CCDFs for early effects. This bias has never been quantified, however. It is an example of uncertainties that arise because of modeling assumptions rather than uncertainties that arise because of poorly known data.

9.6.2.4 Warning Time

In order to assess the benefits of evacuation, it is important to obtain a reliable estimate of the warning time--that is, the interval between the broadcast of a warning and the time of radionuclide release into the atmosphere. This is an important source of uncertainty. It is particularly important for early effects.

9.6.2.5 Particle-Size Distribution

This uncertainty arises because of modeling simplifications and because of a lack of knowledge of the size distribution itself. The most important impact is on deposition modeling (see Appendix D): particle size can account for a difference in the deposition velocity of up to about two orders of magnitude. Hence, any consequence that depends on the deposited level of radioactive material, such as the area of contaminated land, will be subject to uncertainty.

9.6.3 METEOROLOGICAL MODELING

A number of parameters or phenomena to which meteorological modeling is sensitive are shown in Table 9-16, which also evaluates their contributions to uncertainties in CCDFs. Those that are judged to be most important are briefly discussed below.

Table 9-16. Meteorological modeling: sensitivities and uncertainties

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs			Sensitivity studies	Relevant section of this report
		Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage		
Sampling of meteorological data (parameter)	Selection of weather sequences	Major (peaks only)	Low to moderate	Low	Ritchie et al. (1981b)	App. D4.1.2
Number and location of meteorological stations (parameter)	Plume trajectory	Moderate	Low	Low to moderate	Benchmark exercise Zion study[a]	9.4.3
Definition of stability categories (parameter)	Frequency of occurrence of stability categories	Low to moderate	Low	Low to moderate		9.3.1.2
Gaussian theory vs. K-theory (modeling)	Airborne concentrations	Low to moderate	Low to moderate	Low to moderate	Clarke et al. (1979) Nordlund et al. (1980)	App. D1
Choice of dispersion parameters	Airborne concentrations	Moderate to major	Moderate	Moderate	Benchmark exercise Aldrich (1979) Vogt (1981)	9.3.1.3
Changing vs. constant weather (modeling)	Weather sequences	Low[b]	Low[b]	Low[b]	Benchmark exercise McGrath et al. (1977)	9.3.2.4 App. D4
Straight line vs. trajectory vs. multipuff (modeling)	Area covered by plume and evacuation model	Major	Low	Major	Benchmark exercise	App. D4
Inversion lid (parameter)	Height of plume rise	Low	Low	Low	Sprung and Church (1977a)	9.3.1.5
Wind shear (modeling)	Lateral spread of plume	Low	Low	Low	Sprung and Church (1977b)	
Low wind speeds (modeling)	Airborne concentrations	Low	Low	Low		9.3.1.4
Surface roughness (modeling)	Dispersion parameters Deposition velocity	Low to moderate	Low	Low to moderate	Aldrich (1979)	9.3.1.3
Terrain (modeling)	Plume trajectory	Moderate	Low	Moderate	Benchmark exercise	App. D4.3

[a]Commonwealth Edison Company (1981).

[b]The contribution to uncertainties in CCDFs is assessed as low because modelers know which to choose--changing or constant weather.

9.6.3.1 Sampling of Meteorological Data

The sampling of the available meteorological data is discussed in Section D4.1.2. The uncertainties that are attributable to the sampling methods used in CRAC and CRAC2 are shown in Figure 9-17. This illustrates the considerable importance of ensuring that the data are sampled in a reliable way. (See Appendix D4 for further discussion.)

9.6.3.2 Trajectory Versus Straight Line

This uncertainty is discussed in Appendix D (Section D4) and in Section 9.6.2.3.

9.6.4 DEPOSITION

Examples of sensitivities and contributors to uncertainty in the modeling of radionuclide deposition are given in Table 9-17.

9.6.4.1 Dry-Deposition Velocity

As discussed in Appendix D3, important uncertainties arise both in the specification of a value for dry-deposition velocity v_d and in the choice of a deposition model. Kaul (1981b) gives examples of the possible ranges of airborne and deposited concentrations, given a range of values of $v_d/\bar{u}$ (see also Hosker, 1974).

It is pertinent to remark in this context that Beyea (1978a,b) automatically incorporates v_d into his models as an uncertain parameter within a range that varies with stability class as follows:

For stability classes A-D, $0.001 \leq v_d \leq 0.1$ m/sec
For stability class E, $0.001 \leq v_d \leq 0.03$ m/sec
For stability class F, $0.001 \leq v_d \leq 0.01$ m/sec

The Nuclear Power Plant Siting Study performed by Sandia National Laboratories (Strip et al., 1981) contains a sensitivity study of a variation in v_d. This study was carried out for a large, hypothetical release of radioactive material known as SST1.* The "summary evacuation" procedure was assumed,† with a 1120-MWe reactor, the Indian Point population

*Core melt, loss of all safety systems, containment failure and radionuclide release to the atmosphere, and the following release fractions for volatiles: I, 0.45; Cs, 0.67; Te, 0.64. It is the same as the TC-γ' sequence for the rebaselined BWR (USNRC, 1981b).

†Delay times of 1, 3, and 5 hours with respective probabilities of .3, .4, and .3, and an evacuation speed of 10 mph, as described by Aldrich, Blond, and Jones (1978).

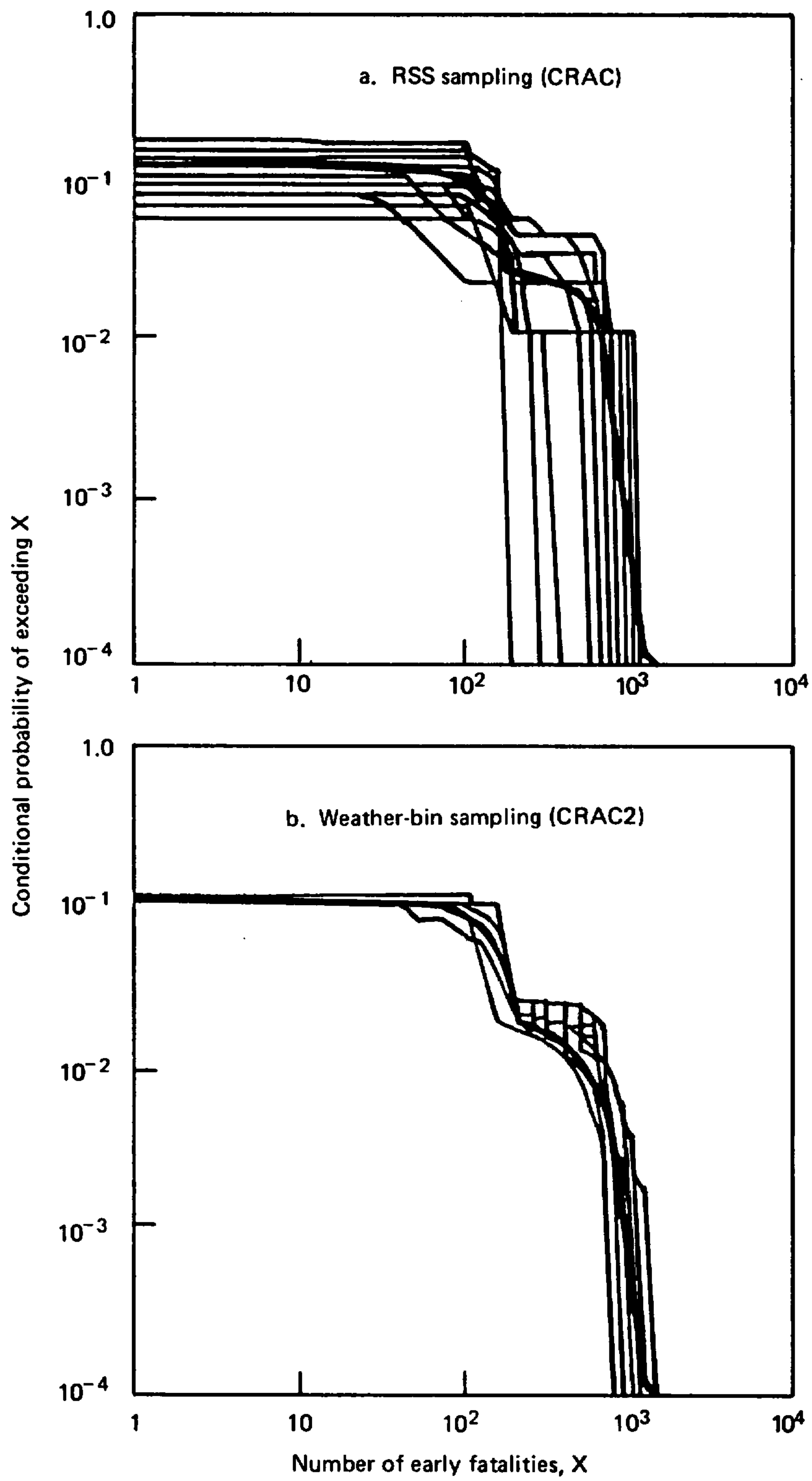


Figure 9-17. Comparison of uncertainty due to sampling by (a) the RSS technique (CRAC) and (b) the weather-bin technique (CRAC2). For each technique, 32 different sets of weather sequences are used to generate early-fatality frequency distributions for a PWR-2 release. A "best estimate," using all 8760 available sequences, is shown by the darkened line. From Ritchie et al. (1981b).

Table 9-17. Deposition modeling: sensitivities and uncertainties

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs: Early fatalities, injuries	Contribution to uncertainties in CCDFs: Latent-cancer fatalities	Contribution to uncertainties in CCDFs: Contaminated area, property damage	Sensitivity studies	Relevant section of this report
Dry-deposition velocity (modeling and parameter)	Quantity of radioactive material on ground	Moderate (major for peaks)	Moderate	Major	Benchmark exercise Beyea (1978a,b) Kaul (1981b)	App. D3
Rainfall model (modeling)	Quantity and location of deposited material	Major (peaks)	Low	Moderate	Ritchie et al. (1976, 1980)	9.3.2.3
Runoff (modeling)	Location of deposited material	Moderate	Moderate	Moderate	Ritchie et al. (1976, 1980)	9.3.2.3

distribution and wind rose, and New York City weather conditions. The deposition velocity (assumed to be the same for all radionuclides except the noble gases) was varied from 0.1 to 10 cm/sec.

Over this range, mean early fatalities varied only by a factor of 1.5. The peak early fatalities varied by a factor of 10, however. Other quantities that are significantly affected by changes in the deposition velocity are the distances within which various effects are predicted to occur: early fatalities, early injuries, and land interdiction (see Table 9-18).

9.6.4.2 Rainfall and Runoff

Section 9.3.2.3 contains a brief description of the rainfall and runoff model of Ritchie et al. (1976). When this new rainstorm and runoff model was used in CRAC, single-trial calculations (one weather sequence containing a rainstorm) yielded probabilities for early and latent fatalities that were increased or decreased by up to an order of magnitude. However, for multiple-trial calculations (91 weather sequences selected from a 1-year record by stratified sampling) mean risk estimates (approximately equal to the area under the CCDFs) were essentially unchanged when compared with the original model, principally because rain is infrequent and therefore consequences produced by weather sequences that contain rain contribute minimally to risk for the nearby public. However, the number of early fatalities predicted for the peak accident is increased because the higher rain rates of the rain cells and the small mesoscale-storm structures cause the levels of deposited radionuclides to be substantially higher over small areas at longer distances, where the chance of encountering large populations is greater.

9.6.5 ACCUMULATION OF RADIATION DOSE

Uncertainties and sensitivities are summarized in Table 9-19. The literature contains far fewer sensitivity studies in this area, but this is not to say that no uncertainties exist. For example, the treatment of weathering is largely based on a single experiment with cesium (Gale et al., 1964). The treatment of ingestion pathways is beset by uncertainties in the calculation of concentration factors, but this has not caused great concern because, for the typical mix of radionuclides that are likely to escape to the atmosphere in the event of an LWR accident, it is generally predicted that ingestion will be only a small contributor to total latent effects (see Tables 9-6 and 9-7). This conclusion would not be true if strontium-90 and cesium-137 were the main components of the release, however.

The next task of the International Benchmark Committee (see page 9-17) will be a thorough survey of chronic effects. This should provide insight into uncertainties in this area.

Table 9-18. Sensitivity of the distances to which consequences occur for various deposition velocities[a]

Dry-deposition velocity (cm/sec)	Distance (miles)											
	Early fatalities				Early injuries				Land interdiction			
	Mean	90%	99%	Peak calculated	Mean	90%	99%	Peak calculated	Mean	90%	99%	Peak calculated
0.1	2.1	4	15	25	7.2	15	55	65	11	30	60	100
0.3	1.9	4	15	25	7.1	20	40	50	16	40	65	85
1.0	1.7	4	12	18	8.3	25	35	50	19	40	60	35
3.0	1.6	3	4	18	6.6	12	23	25	20	25	40	45
10.0	1.4	3	3	3	3.5	6	15	18	13	22	23	25

[a]From Strip et al. (1981).

Table 9-19. Accumulation of radiation dose: sensitivities and uncertainties

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs: Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage	Sensitivity studies	Relevant section of this report
Dose-conversion factors (modeling)	Radiation dose delivered by inhalation	Low[a]	Low[a]	Low[a]	Hunt et al. (1979) Kelly et al. (1979)	9.3.3.1
Methods of approximating cloudshine and groundshine (modeling)	Computing time Externally delivered radiation dose	Low to moderate	Low to moderate	Low to moderate	Van der Hoven and Gammill (1979)	9.3.3.2
Resuspension (modeling)	Long-term inhaled radiation dose	Low	Low	Low	Wall et al. (1977)	9.3.3.4
Ingestion-pathway modeling	Chronic radiation dose	Zero	Moderate	Moderate		9.3.3.3
Weathering (modeling)	Long-term externally delivered radiation dose	Low	Moderate	Major		9.3.3.2

[a]For light-water reactors.

9.6.6 MEASURES THAT CAN REDUCE PREDICTED RADIATION DOSES

Uncertainties and sensitivities in the effects of preventive countermeasures are summarized in Table 9-20. In general, the uncertainties in the evacuation model produce large uncertainties in the corresponding CCDF for early fatalities. The uncertainties in the interdiction and decontamination countermeasures lead to the greatest uncertainties in contaminated areas and property damage.

9.6.6.1 Delay Time in Evacuation Model

Delay time in evacuation is one of the most important parameters* in the consequence model, as can be seen from Figure 9-7, from Aldrich, Ritchie, and Sprung (1979). The reader is referred to the discussion in Appendix E, Sections E1 and E4.

9.6.7 HEALTH EFFECTS

Sensitivities and uncertainties in the predicted health effects are summarized in Table 9-21. This is another area in which there is a relative paucity of sensitivity studies.

9.6.7.1 Dose-Response Relationships: Thresholds

Early fatalities and early injuries are threshold effects. Clearly, assigning a threshold to a dose-risk relationship like that for early fatalities can make an important impact on the number of people who are affected at radiation-dose levels above that threshold.

9.6.7.2 Medical Treatment

Figure VI F-1 of Appendix VI of the Reactor Safety Study (USNRC, 1975) gives three dose-response curves for early mortality, assuming (1) minimal medical treatment, (2) supportive medical treatment, and (3) heroic medical treatment. The thresholds for these relationships are 150, 220, and 890 rads, respectively, and the 50-percent probability levels are at about 320, 510, and 1000 rem, respectively. Each of these dose-response curves would lead to a very wide range of the CCDFs predicted for early effects if they were used in turn in a consequence model.

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*More precisely, it is the difference between the warning time and the delay time that is important.

Table 9-20. Preventive countermeasures: sensitivities and uncertainties

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs			Sensitivity studies	Relevant section of this report
		Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage		
Delay time and warning time in evacuation model (modeling and parameter)	Short-term radiation dose	Major (except for peaks)	Low	Low	Aldrich, Ritchie, and Sprung (1979)	App. E1
Effective evacuation speed (parameter)	Short-term radiation dose	Moderate (low for peaks)	Low	Low	Aldrich, Ritchie, and Sprung (1979)	App. E1
Radial vs. realistic evacuation routes	Short-term radiation dose	Moderate (low for peaks)	Low	Low		App. E3
Other evacuation parameters (e.g., radius of evacuation zone)	Externally delivered radiation dose	Moderate	Low	Low	Aldrich, Ritchie, and Sprung (1979)	App. E1
Sheltering strategies and shielding factors (parameters)	Externally delivered radiation dose	Moderate (major for peaks)	Low	Low	Aldrich, Ritchie, and Sprung (1979)	App. E2
Interdiction (parameter)	Number of people relocated	Low	Moderate	Moderate to major		9.3.4.3
Decontamination (parameter)	Area of land interdicted, costs	Low	Moderate	Major		9.3.4.4
Thyroid blocking; respiratory protection; ventilation strategies (modeling)	Inhaled radiation dose	Low[a]	Low[a]	Low[a]	Aldrich and Blond (1980, 1981) Aldrich and Ericson (1977)	9.3.4.5

[a]Not usually incorporated into consequence models.

Table 9-21. Health effects: sensitivities and uncertainties

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs			Sensitivity studies	Relevant section of this report
		Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage		
Dose-response relationships for early effects (modeling)	Probability of occurrence of health effect	Moderate to major	Low	Low		9.3.5
Age-cohort treatment (modeling)	Inhalation dose-conversion factors and dose-response relationships	Low	Low	Low		9.3.5 9.3.3.1
Time intervals over which dose accumulates (modeling)	Dose used in dose-response relationship	Moderate	Moderate	Low		9.4.8
Dose-response relationships for cancer induction (modeling)	Probability of latent health effects	Low	Major	Low	Beyea (1978a,b) BEIR III (NAS-NRC, 1980)	9.3.5 9.4.8.4

9.6.7.3 Linear or Other Hypothesis for Cancer Induction

In the BEIR III report (NAS-NRC, 1980), estimates of the probability of cancer induction vary over an order of magnitude for low doses and dose rates. Beyea (1978a,b) accounts for this by taking the number of fatal cancers to be between 50 and 500 per 10^6 whole-body man-rem. Clearly, this could lead to an uncertainty of as much as an order of magnitude on the CCDFs for latent-cancer fatalities.

9.6.8 PROPERTY DAMAGE AND ECONOMIC COSTS

Sensitivities in the modeling of property damage and economic costs are shown in Table 9-22, which also ranks their contributions to uncertainties in CCDFs. Very few sensitivity studies, if any, have been done to estimate the width of the error bands on CCDFs for areas of interdicted land or economic costs. Possibly, further work should be carried out in this area.

9.6.9 DEMOGRAPHIC DATA

Sensitivities and uncertainties related to demographic data are displayed in Table 9-23. As discussed in Section 9.4.4, one of the most difficult problems is the treatment of diurnal variations in populations as people move between work and home. This affects not only the movement of people in relation to the plume but also evacuation and shielding strategies.

9.6.10 DISCUSSION

The uncertainties discussed above remain, for the most part, unquantified. Those studies that have attempted to quantify them have done so in a subjective manner. For example, in the Zion study (Commonwealth Edison Company, 1981), all of the uncertainties in the consequence model were simulated by a judgmental probability distribution on the calculated radiation doses. If the dose initially calculated for the best estimate was of magnitude Q, it was judged that the uncertainties could be represented by the following: (1) a probability of .1 that the dose has magnitude 2Q; (2) a probability of .35 that the dose has magnitude Q; (3) a probability of .45 that the dose has magnitude 0.5Q; and (4) a probability of .1 that the dose has magnitude 0.1Q. It is to be stressed that nobody has yet done better than this simple approach. Hence, the use of sophisticated Bayesian or classical techniques to quantify uncertainties in consequence analysis has not yet been attempted.

Future research into uncertainties in consequence modeling may well be directed into two separate channels. The first will be to reduce uncertainties in some of the important parameters. An example of this could be to take advantage of current interest in radionuclide source terms, which

Table 9-22. Property damage and economic costs: sensitivities and uncertainties

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs: Early fatalities, injuries	Contribution to uncertainties in CCDFs: Latent-cancer fatalities	Contribution to uncertainties in CCDFs: Contaminated area, property damage	Sensitivity studies	Relevant section of this report
Decontamination effectiveness (parameter)	Area of interdicted land	Low	Low to moderate	Moderate to major		9.3.4.4
Runoff and weathering (modeling)	Area of interdicted land	Low to moderate	Low to moderate	Moderate	Ritchie et al. (1976)	9.3.3.2 9.3.2.3
Cost elements in economic model (parameters)	Costs	Zero	Zero	Moderate		9.4.7

Table 9-23. Demographic data: sensitivities and uncertainties[a]

Parameter or modeling assumption	Quantity most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs			Relevant section of this report
		Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage	
Transient population variations	Total number of health effects	Moderate, but peaks could be major (site dependent)	Low to moderate	Low to moderate	9.4.4
Diurnal variations	Total number of health effects	Moderate to major (site dependent)	Low	Low to moderate	9.4.4
Assigning population to elements of grid	Total number of health effects	Low to moderate	Low	Low	9.4.4

[a]No sensitivity studies are known to the authors.

should lead to improved methods for predicting the magnitudes of the source terms and to an improved specification of the properties of particulate radionuclides released into the atmosphere; this ought to lead, in turn, to improved deposition modeling. The second channel will be to devise more sophisticated methods of estimating uncertainties, taking into account correlations between parameters and models.

9.7 DOCUMENTATION

This section presents a suggested outline for a final report on a consequence-modeling exercise or for a chapter on consequence modeling in the final report summarizing a complete PRA exercise. The consequence modeler is not obliged to follow this outline if he has one that he thinks is better. The main requirement is that the reader be able to follow what has been done and, in particular, be able to understand the various input parameters.

9.7.1 INTRODUCTION

The introduction should begin by stating the reason for undertaking the consequence analysis. It should then go on to say which consequence-modeling code was used and why.

9.7.2 METHODS

This section should begin by describing the consequence-modeling code that was chosen. Obviously, there is no need to duplicate the user's guide to the code or to reproduce vast portions of the Reactor Safety Study. A brief description is all that is required, with copious references if need be.

The most important part of this section should be the description of what changes, if any, have been made to the chosen code, with the reasons for making these changes. This description is necessary in order to assist in the interpretation of the final results and to answer such inevitable questions as, Why does this differ from the Reactor Safety Study?

9.7.3 INPUT DATA

The gathering and processing of input data, which were discussed at some length in Section 9.4, are the activities with which the user of consequence models can most influence the output. It is essential, therefore, that he write down clearly what he has done. This section should consist of a review of each piece or set of input data, including--

1. The source of the data, properly referenced.

2. How the data were processed.

3. If relevant, what has been done to overcome problems associated with data of poor quality (this has been mentioned particularly in the context of meteorological data).

4. Tabulation of the actual values used for input parameters, if possible.

As described in Section 9.4, the areas in which collections of input data are required include the following:

1. Source-term specification.
2. Meteorological data.
3. Population data.
4. Deposition input.
5. Economic data.
6. Health-physics data (dosimetry and health effects).
7. Evacuation data.
8. Basic radionuclide data.

Depending on the code used, other groups of data may be required. The importance of writing this section clearly and comprehensively cannot be overestimated. Without it, the reader of the final report will have grave difficulty in understanding the results.

If more than one run of the code is carried out, perhaps in order to assess sensitivities to some parameters, the range of inputs used should also be specified.

9.7.4 RESULTS AND INTERPRETATION

This section should contain the results of the study. Presumably there will be some base case that has been calculated with best estimates of all input parameters and what is thought to be the best available set of models. The output for this base case should be presented as the final results of the study; the sort of output that is possible is discussed in Section 9.5.

This set of results should be accompanied or followed by interpretation. This interpretation may include the sensitivity to various input parameters--which might be reactor-oriented (possible variations in dominant accident sequences, say) or ranges of values in a parameter like the dry-deposition velocity--or to modeling changes made by the user of the code.

The interpretation of the results may also include comparisons with the results of other studies. For example, it may be thought desirable to compare calculated CCDFs with those given in the Reactor Safety Study, in order to explain what improvements have been brought about by changes in modeling and parameters in the consequence code, or by improved event- and fault-tree analyses of the dominant accident sequences, or by changes in the design of the reactor itself. It may also be desirable to compare estimated public risks with other man-caused risks.

The section should include a discussion of the uncertainties associated with the results. As explained in Section 9.6, it will not be possible to quantify these uncertainties in any exact statistical sense. Nonetheless, they exist and cannot be ignored.

Finally, this section should contain the conclusion or conclusions of the study, clearly stated. Examples could be that the predicted risk to the public from the operation of the reactor in question has been shown to be very small or that a certain design change has been very effective in reducing public risk. The nature of the conclusion clearly depends to some extent on the purpose of the study and, as has been shown in Section 9.1.2, there are a variety of purposes for a consequence analysis.

9.7.5 MISCELLANEOUS

There will probably be a need for a final section giving information on miscellaneous items such as the assurance of technical quality, acknowledgments, and references.

9.8 ASSURANCE OF TECHNICAL QUALITY

There are two aspects to the assurance of technical quality. The first is to ensure that the user has obtained a reliable code in good working order. The second is to ensure that he has used the code in a proper manner. Nowadays, a typical consequence-modeling code is such a vast compilation of multidisciplinary models in various fields--the modeling of turbulent processes such as meteorology and buoyant plume rise, health physics, economics, social studies, biology, soil physics, etc.--that no user can hope to go through it module by module, understand every detail, and verify that it is working as claimed in the user's guide. Indeed, there is a sense in which no consequence-modeling code has been properly validated. Such codes purport to predict the consequences of large accidental releases of radioactivity into the atmosphere and to predict the number of early and latent fatalities that might occur in the surrounding population. Since there has never been an accidental release of radioactivity from a commercial reactor that has caused the death of any member of the public, or indeed any detectable injury, the predictions of such codes cannot be validated, and it is hoped that they never will be.

The code user therefore has to rely on the presumed competence of the code developer. He has to assume that the models incorporated into the various modules of the code--the meteorological model, for example--represent good practice. He should read the code manual and associated literature with a critical eye, noting the sources cited for the various models and the justifications given for their use. He should not hesitate to query the code developer if there is any reason for doubt. This is another reason why the first task of the user, described in Section 9.2.1, namely, acquiring background, is so important. Consequence models should never be used as a black box.

What the user must do is to make sure that, once he has obtained a code, it is running properly on his machine. To do this he requires samples of input and output from the code developer. These samples should cover all areas of the code that he is likely to use in his consequence analysis.

If the user alters the code he has obtained, to incorporate better modeling, he should describe the reasons for the alterations and reference them. He should also carry out in-house calculations to ascertain that the new modeling is working correctly.

If, after this, the user is still unhappy about his code, there should shortly exist a means whereby he can check the predictions of the code by carrying out standard calculations and comparing the results with those of other consequence-modeling codes. This is the Benchmark exercise, already mentioned in Section 9.2.3. The forthcoming report of this exercise will specify seven standard problems designed to exercise most parts of consequence-modeling codes. The results obtained by some 20 organizations in Europe, the United States, and Japan will also be published. The user of a consequence model should repeat these standard runs and see whether his results lie within the envelope generated by a worldwide community of consequence-modeling experts. If he does, well and good. If not, he must examine his code to determine why his results differ. If he wishes to stand by his results, he must know his code well enough to determine the reasons for the difference and to justify the modeling or parameters that cause the difference.

Once the code is put to use in a specific consequence analysis, the problem of ensuring technical quality reduces to that of justifying the input data used. This can be done by compiling the data as described in Section 9.4 and by describing it clearly as outlined in Section 9.7.3. The input data set should also be independently reviewed in order to make sure that the collected data were actually input to the code. Finally, the output of the calculation should be presented and interpreted as described in Section 9.5.

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Chapter 10

Analysis of External Events

10.1 INTRODUCTION

External events (e.g., earthquakes, floods, fires, tornadoes, aircraft impacts, and explosions) can be considered at any level of PRA, depending on the objectives and the scope of the study, as discussed in Chapter 2. This present chapter describes how external events are selected for detailed evaluation in a PRA, discusses the methods used to evaluate their hazards, and explains how the assessment of external events is integrated with the analysis of internal events in evaluating the total plant risks. An overall procedure for treating external events is discussed here; details specific to some particular external events are presented in Chapter 11.

It should be noted that the basic PRA methods and procedures presented in the preceding chapters are generally applicable to all risk contributors, including the so-called external events. However, there are valid reasons for setting aside separate chapters of this procedures guide to discuss the analysis of external events. Most important, the analysis of external events requires the use of specialized methods to address important factors not usually encountered in the analysis of the internal events. These include the assessment of frequency of occurrence versus magnitude for external events and the modeling of component and structure failure in terms of variables that describe physical interactions. Since a complete risk analysis of an external event would entail many of the PRA elements generic to any risk contributor, there is a motivation to modularize the steps in the risk-analysis procedure for external events so as to avoid overlaps. The interfaces between external event analysis and the basic event- and fault-tree logic are discussed in Section 10.3.6.

In addition to natural and man-induced external events, the scope of this chapter includes internal flooding, fire, and turbine missiles, which are not external events in the strict sense. Events like sabotage and war are not included, although it is recognized that, with appropriate refinements, the overall procedure is applicable to these events also.

In specific application, the risk-assessment methods described here have heretofore been limited to earthquakes, winds, fires, and floods. Of the methods described, some are common to all external events and others are specific to a particular event. Application to other events not yet analyzed in detail may therefore require some additional development. Furthermore, it is recognized that the degree of uncertainty in estimating risk due to accidents caused by external events tends to be greater than that associated with other accident-initiating events that have been analyzed. Greater uncertainties stem from less experience in analyzing external events, lack of data, the use of relatively new analytical techniques, and greater reliance, perhaps, on engineering judgment and expert opinion. Engineering judgment is, however, not intended to replace the concerted effort to quantify the external events. If external events are deemed to significantly contribute to the overall plant risk, work on analytical models and data collection may be encouraged in the future, and its results may eventually reduce the uncertainties that now must be assigned. In the meanwhile, a detailed peer review of the assumptions, models, and input-parameter values is necessary to achieve consistency between different PRA studies relying heavily on engineering judgment in the treatment of external events.

The greater degree of uncertainty should not be construed as a reason for excluding external events in a PRA. Indeed, assessment of the magnitude of the effects of uncertainty is a key component of the risk quantification. Consistent with the overall philosophy of PRA, the analyst has to develop a complete description of each external event phenomenon and its effect on plant risk. Such a description should include not only the best estimate of the contribution of external events to plant risk but also the uncertainty in that contribution. Hence, greater uncertainty results in wider error bands about the best estimate of plant risk.

The selection of any external event for a detailed risk analysis will depend on its frequency of occurrence, magnitude, proximity, and consequences. The results of the external event analysis will be used as input to the PRA in defining initiating events, in developing event and fault trees for accident-sequence and system analysis, and in quantifying accident sequences. The depth of analysis suggested here for external events is commensurate with the overall objectives of this document, and the procedures presented represent the current state of the art in analyzing risks from external events. Since they are still in a developmental stage, it can be expected that the methods used in the analysis of external events will undergo significant changes as the industry gains experience in the treatment of external events in PRA studies.

In considering external events, the PRA analyst faces two fundamentally different types of variability. One is fundamental to the phenomenon being represented; the other is incomplete knowledge about the representation of that fundamental variability. Throughout Chapters 10 and 11, the word "frequency" is used when the inherent randomness of variables and events is discussed, and "probability" is used to refer to the uncertainty or current level of ignorance concerning the variables and events. In order to maintain this distinction and to treat both kinds of variability consistently, the "probability-of-frequency" format proposed by Kaplan and Garrick (1981) is adopted in the discussion of external events. More details on this format can be obtained from Kaplan et al. (1981).

10.2 OVERVIEW

10.2.1 SELECTION OF EXTERNAL EVENTS

The PRA studies that have been conducted to date have treated external events to varying degrees of detail. Some studies have excluded these events altogether. Some other studies have been motivated by external events (Pacific Gas & Electric, 1977; Smith et al., 1981). Since at present the collective experience of the industry in performing PRAs for nuclear power plants is rather limited and until sufficient sensitivity studies are conducted to assess their relative contribution to plant risk, external events cannot be dismissed a priori. Hence, a formal procedure is needed to ensure that all potential external events are considered and that the significant ones are selected for detailed PRA studies.

Although a detailed risk assessment is performed only for a few selected events, it is to be understood that the final plant-risk estimate includes the contributions from all external events. The contributions from events considered insignificant may be too small to show up in the significant digits reported for the total risk estimate. For example, let us assume that the mean (or best estimate) frequency of a particular release category for earthquakes is 10^{-6} per year. If the mean frequency of the same release category for aircraft impact is calculated as 10^{-8} per year, the mean release frequency from these two events is approximately 1.01×10^{-6} per year. The analyst may choose to report this estimate as 1×10^{-6} per year, thereby masking the contribution from the aircraft impact.

The screening of external events to select the significant ones consists of several steps. First, all external events specific to the site and plant are identified (Table 10-1 on pages 10-8 and 10-9 should be reviewed to ensure that all external events are indeed considered). Screening criteria are then established. Using these criteria, each external event is reviewed to judge whether it deserves further study. The external events that are discarded as being insignificant should be documented in the PRA study report along with the reasons for not performing a detailed analysis.

10.2.2 ASSESSMENT OF RISKS FROM EXTERNAL EVENTS

As shown in Figure 10-1, the basic elements of the analysis of risk from an external event are (1) hazard analysis, (2) plant-system and structure response analysis, (3) evaluation of the fragility and vulnerability of components (structures, piping, and equipment), (4) plant-system and sequence analysis, and (5) consequence analysis. The outputs are release or damage-state frequencies and risks. The information developed in hazard analysis, response analysis, and in component-fragility evaluation is input into the overall system models described in Chapter 3 and appropriately modified for the external event under study. The accident sequences specific to this external event are then quantified, and the frequencies of different release categories are calculated. The consequence analysis discussed in Chapter 9 is performed by using a consequence-analysis model that reflects the effects of an external event on the environment (e.g., a large earthquake or a severe flood may disrupt the communications network and damage evacuation routes, so that the distribution of the population exposed to radiation may be different than that for internal events). The plant risk, expressed as a frequency of exceedence (complementary cumulative distribution function) of damage (e.g., early fatalities, latent-cancer fatalities, or property damage), is calculated using the procedures discussed in Chapter 9. At each stage of the external event analysis, the analyst(s) should quantify the uncertainty in the output (e.g., uncertainty in the frequency of exceeding different levels of hazard intensity, uncertainty in component fragilities), and these uncertainties should be appropriately propagated through the entire analysis.

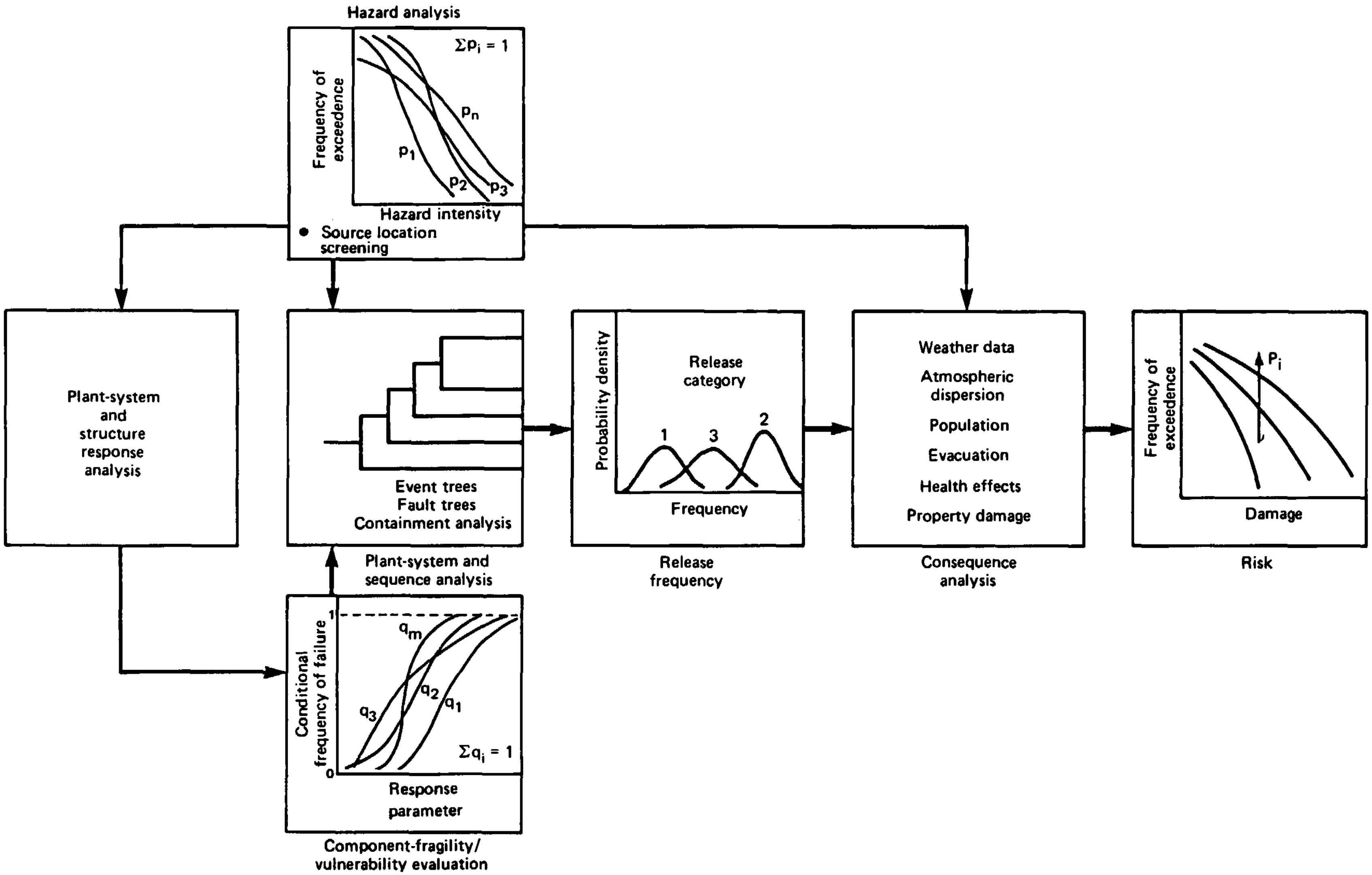


Figure 10-1. Risk-assessment procedure for external events.

A detailed risk analysis may not be warranted for some external events. The hazard-analysis results, combined with some bounding evaluation of plant damage and consequences, may indicate that the risk contribution of the external event is smaller by several orders of magnitude than those of other internal and external events. For example, the PRA study team may calculate the mean frequency of aircraft-impact damage to any one of the vulnerable structures whose failure could lead to core melt. If this frequency is much lower (e.g., 1/100) than those for other events and if the consequences of core melt from aircraft impact are comparable to other events in likelihood and magnitude, then no further detailed risk assessment for aircraft impact may be necessary.

In hazard analysis, the frequencies of occurrence of different intensities of an external event, called "hazard intensities," are calculated and presented in the form of a hazard curve. The hazard intensities could be represented by the peak ground acceleration of earthquakes, tornado intensities measured on the Fujita-Pearson intensity scale (Fujita, 1971), the sizes (weights) of aircraft, or the overspeed conditions under which turbine missiles are generated. Characterizing a complex hazard phenomenon by such a single parameter is generally inadequate. However, the other parameters that could be used to completely describe a hazard (e.g., for earthquakes, they could be duration, frequency content, etc.) are used in defining the hazard input to the response analysis. Also, the particular parameter selected to characterize an external event depends on the plant-system and sequence analysis. If the initiating events in the system event trees are related to different levels of earthquakes (e.g., 0.20g, 0.30g, and 0.40g), then the parameter of interest is the peak ground acceleration. If the initiating event is a fire in a specific area of the plant, the hazard analysis may consist of evaluating the mean rates of occurrence of fires of different sizes in various areas of the plant. The uncertainties in the hazard-parameter values and in the mathematical model of the hazard are represented by developing a family of hazard curves; a probability is assigned to each hazard curve. The summation of probabilities assigned over the family of hazard curves is unity.

In the response analysis, the response of plant systems and structures for a specified hazard input is calculated. The response of interest is generally the structural response at selected structural, piping, and equipment locations. For earthquakes, the response parameters could be spectral acceleration, moment, and deflection. For extreme winds, they could be force or moment on a structural element and deflection. For some external events (e.g., fire), no specific response analysis is performed.

In the evaluation of component fragility and vulnerability, the conditional frequencies of component failure for different values of the response parameter are calculated. Again, some differences exist between external events, depending on the plant-system and sequence analysis. For example, in a seismic risk analysis, fragilities may be expressed as functions of the local response parameter and evaluated separately for each component. In an analysis of risks from turbine missiles, the conditional frequencies of failure from turbine-missile impact are evaluated for different components in an accident sequence. These frequencies of failure depend on the location of the component with respect to the missile trajectory; the missile ricochet effects and the structural capacity of barriers are considered in calculating these frequencies. The uncertainties in the component-fragility parameters and the mathematical model are represented by developing a family of fragility curves for each component; a probability is assigned to each fragility curve. The summation of probabilities assigned over the family of fragility curves is unity. For external events that have discrete hazard-parameter values (e.g., turbine overspeeds and aircraft sizes), the component fragility is calculated at the corresponding discrete response values. The uncertainties are expressed by assigning probabilities to a vector of fragility values for a specified response value.

The plant-system and sequence analysis is performed by developing event trees and fault trees with an external event of a particular hazard intensity as the initiating event. The component fragilities are then used to compute the frequencies of failure for different safety systems. A very important consideration here is the dependences or correlations involved in the assignment of frequencies to multiple component failures. The calculated failure frequencies are conditional on the specified hazard intensity. The unconditional frequency of core melt or of radionuclide release for a given release category is obtained by integrating over the entire range of hazard intensities.

It may be logical to merge the external event analysis with the internal event analysis at the stage of plant-system and sequence analysis. In such an approach, the systems analysts should be apprised of the particular features of the external event that differ from the internal events. These features include, but are not limited to, differences in initiating events, in the event and fault trees, in the containment event trees, and in the quantification of accident sequences. The details of this interface are discussed in Section 10.3.6.

In some recent PRA studies, however, the analysts have chosen to treat the external events separately and to calculate the frequencies of release categories resulting from the external events. Several advantages are claimed for this treatment: (1) the differences of the external event analysis (e.g., initiating events, event and fault trees, containment-failure modes, and the quantification of fault trees) are made highly visible, resulting in the development of special analytical techniques; (2) the release-frequency analysis can be carried out with simplified plant-level fault trees; (3) the dependences between component failures that result from correlations between responses arising from the same loading (external event) and between component capacities arising from a common vendor and similar mounting can be handled explicitly; and (4) the contributions of different external events to core-melt frequency, release frequencies, and damage frequencies can be studied with a view to identifying the dominant events and planning optimal strategies for reducing (if needed) plant risk. The results of the external event analysis, in the form of frequencies of release categories, are then used, along with similar information from the internal event analysis, as input to the consequence analysis, if the analyst considers the differences between external and internal events in the consequence analysis to be insignificant; otherwise, the consequence analysis is carried out separately. The final product of the external event analysis is then an estimate of plant risk. The recently published Zion study (Commonwealth Edison Company, 1981) provides examples of how risks from external events are calculated.

The external event analysis should address the influence of design and construction errors and human errors due to operator action or inaction. In the PRA studies performed to date, design and construction errors have been partly accounted for by using as-built drawings and by visually inspecting existing plant conditions in a "walk-through" of the plant. Component-fragility evaluations have considered the customary tolerances in construction and manufacturing. The random equipment failures considered in these studies have included unavailability due to maintenance errors. Operator action in mitigating an accident may not be effective under extreme stress conditions (e.g., beams and walls cracking and collapsing in the control room under a large earthquake or a major fire in the control room). Operator errors of commission (e.g., turning off a wrong valve) under extreme stress were, however, not included in the past studies. The question of design and construction errors and human response deserves further study, as mentioned later in Chapter 13.

10.3 METHODS AND PROCEDURES

10.3.1 IDENTIFICATION AND SELECTION OF EXTERNAL EVENTS

An extensive review of information on the site region and plant design should be made to identify all external events to be considered. The data in the safety analysis report on the geologic, seismologic, hydrologic, and meteorological characteristics of the site region as well as present and projected industrial activities (i.e., the building of a reservoir, the construction of a road that carries hazardous materials, increases in the number of flights at an airport, etc.) in the vicinity of the plant should be reviewed for this purpose. The list of external events is to be exhaustive and is not to be constrained by any limitations on size or intensity; the screening techniques are meant to identify the significant external events to be included in the detailed risk assessment.

Table 10-1 lists the natural and man-made external events that should be considered in a PRA study. This list should be reviewed by the PRA study team to ensure that all applicable external events are included in the risk assessment. Although every attempt was made to list all possible external events, Table 10-1 should not be treated as an exhaustive set.

The external events identified as described above are screened in order to select the significant events for a detailed risk quantification. The PRA study team should formulate a set of screening criteria that should minimize the possibility of omitting significant risk contributors while reducing the amount of analysis to manageable proportions. As an example, a set of screening criteria is given below. Each of these criteria provides an acceptable basis for excluding external events from a detailed risk assessment. An external event is excluded if--

1. The event is of equal or lesser damage potential than the events for which the plant has been designed. This requires an evaluation of plant design bases in order to estimate resistance to a

Table 10-1. Natural and man-induced external events to be considered in PRA studies[a]

Event	Applicable screening criterion[b]	Remarks
Aircraft impact	--	Site specific; requires detailed study
Avalanche	3	Can be excluded for most sites in the United States
Coastal erosion	4	Included in the effects of external flooding
Drought	1	Excluded under the assumption that there are multiple sources of ultimate heat sink or that the ultimate heat sink is not affected by drought (e.g., cooling tower with adequately sized basin)
External flooding	--	Site specific; requires detailed study
Extreme winds and tornadoes	--	Site specific; requires detailed study
Fire	--	Plant specific; requires detailed study
Fog	1	Could, however, increase the frequency of man-made hazard involving surface vehicles or aircraft; accident data include the effects of fog
Forest fire	1	Fire cannot propagate to the site because the site is cleared; plant design and fire-protection provisions are adequate to mitigate the effects
Frost	1	Snow and ice govern
Hail	1	Other missiles govern
High tide, high lake level, or high river stage	4	Included under external flooding
High summer temperature	1	Ultimate heat sink is designed for at least 30 days of operation, taking into account evaporation, drift, seepage, and other water-loss mechanisms
Hurricane	4	Included under external flooding; wind forces are covered under extreme winds and tornadoes
Ice cover	1, 4	Ice blockage of river included in flood; loss of cooling-water flow is considered in plant design
Industrial or military facility accident	--	Site specific; requires detailed study
Internal flooding	--	Plant specific; requires detailed study
Landslide	3	Can be excluded for most sites in the United States
Lightning	1	Considered in plant design
Low lake or river water level	1	Ultimate heat sink is designed for at least 30 days of operation, taking into account evaporation, drift, seepage, and other water-loss mechanisms

Table 10-1. Natural and man-induced external events to be considered in PRA studies[a] (continued)

Event	Applicable screening criterion[b]	Remarks
Low winter temperature	1	Thermal stresses and embrittlement are insignificant or covered by design codes and standards for plant design; generally, there is adequate warning of icing on the ultimate heat sink so that remedial action can be taken
Meteorite	2	All sites have approximately the same frequency of occurrence
Pipeline accident (gas, etc.)	--	Site specific; requires detailed study
Intense precipitation	4	Included under external and internal flooding
Release of chemicals in onsite storage	--	Plant specific; requires detailed study
River diversion	1, 4	Considered in the evaluation of the ultimate heat sink; should diversion become a hazard, adequate storage is provided
Sandstorm	1	Included under tornadoes and winds; potential blockage of air intakes with particulate matter is generally considered in plant design
Seiche	4	Included under external flooding
Seismic activity	--	Site specific; requires detailed study
Snow	1, 4	Plant designed for higher loading; snow melt causing river flooding is included under external flooding
Soil shrink-swell consolidation	1	Site-suitability evaluation and site development for the plant are designed to preclude the effects of this hazard
Storm surge	4	Included under external flooding
Transportation accidents	--	Site specific; require detailed study
Tsunami	4	Included under external flooding and seismic events
Toxic gas	4	Site specific; requires detailed study
Turbine-generated missile	--	Plant specific; requires detailed study
Volcanic activity	3	Can be excluded for most sites in the United States
Waves	4	Included under external flooding

[a]Modified from ANSI/ANS-2.12-1978 (American Nuclear Society, 1978).

[b]See Section 10.3.3 for a sample set of screening criteria. The values given in this table are intended for illustration purposes only. For a specific PRA project, the analyst of external events should establish site-specific screening criteria and apply them to select the external events that may require a detailed study.

particular external event. For example, it is established that safety-related structures designed for earthquake and tornado loadings can safely withstand a 1-psi peak positive incident overpressure from explosions (USNRC, 1978). Hence, if the PRA analyst demonstrates that the overpressure resulting from explosions at a source (e.g., railroad, highway, or industrial facility) cannot exceed 1 psi, these postulated explosions need not be considered. It is assumed that the conditional frequencies of failure of structures and components for overpressures of less than 1 psi are negligible given that the safety-related structures are designed for earthquake and tornado loadings. This screening criterion is not applicable to events like earthquakes, floods, and extreme winds since their hazard intensities could conceivably exceed the plant design bases.

2. The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events. For example, the PRA analyst may exclude an event whose mean frequency of occurrence is less than some small fraction of those for other events; the uncertainty in the frequency estimate for the excluded event is judged by the PRA analyst as not significantly influencing the total risk. Alternatively, the analyst may decide to compare event occurrence frequencies at some high confidence level (e.g., 95 percent). After the total plant risk is estimated, the deleted external events may have to be reviewed to ascertain that a detailed assessment would not reveal them as significant contributors to the total plant risk.

3. The event cannot occur close enough to the plant to affect it. This is also a function of the magnitude of the event. Examples of such events are landslides, volcanic eruptions, earthquake-fault ruptures (seismic motion and its effects are treated under seismic events), and explosions.

4. The event is included in the definition of another event. For example, storm surges and seiches are included in external flooding; the release of toxic gases from sources external to the plant is included in the effects of either pipeline accidents, industrial or military facility accidents, or transportation accidents.

For the sake of illustration, the above screening criteria are applied in Table 10-1 to indicate which external events may be deleted from a detailed risk assessment. It is seen that a detailed risk assessment is required for the following external events:

1. Aircraft impacts.
2. External flooding.
3. Extreme winds and tornadoes (and associated missiles).
4. Fire.
5. Accidents in nearby industrial or military facilities.
6. Internal flooding.
7. Pipeline accidents (gas, etc.).

8. Release of chemicals stored at the site.
9. Seismic events.
10. Transportation accidents.
11. Turbine-generated missiles.

The analyst is urged to use caution in applying screening criteria based solely on plant design bases. Some design-basis arguments were made to justify the exclusion of external events from early PRA studies. However, recent attempts at quantifying the risk associated with external events have led to some surprising results, as described in Chapter 11. One of the pitfalls of using criteria based on plant design bases is that emphasis is placed on comparisons of the event list with the design bases of the so-called safety-related systems and structures. However, there are important risk contributions from the so-called non-safety-related systems as well, and their capabilities and capacities are neither well defined nor documented. Moreover, the analyst should not lose sight of the possibility that the magnitude of an external event may exceed the plant design basis. It may be necessary to perform some bounding estimates of the risk contribution before a convincing case can be made for excluding any given external event from a PRA.

The screening of external events may start at the beginning of a PRA project in order to maintain the schedule. However, once an external event has been identified for a detailed PRA study, it is more efficient to perform the task after the event and fault trees for the internal events are developed so that the consequences of failures are more apparent.

The next section describes how a detailed risk assessment for these selected external events can be performed.

10.3.2 METHOD FOR ASSESSING RISKS FROM EXTERNAL EVENTS

As already mentioned, the basic elements of the analysis of risk from an external event are (1) hazard analysis, (2) plant-system and structure response analysis, (3) evaluation of component fragility and vulnerability, (4) plant-system and sequence analysis, and (5) consequence analysis. Depending on the stage at which the analyses of internal and external events are merged together, the final product of the external event analysis could be (1) results of hazard analysis, component fragilities, and modifications to system event and fault trees, and containment-failure modes; (2) probability distributions of frequencies for various release categories; or (3) probability distributions of frequencies for various damage indices (e.g., early fatalities, latent-cancer fatalities, or property damage).

The PRA of an external event can be viewed as a problem in determining $f_k(z)$, the unconditional frequency of exceeding damage level z of consequence type k, resulting from potential reactor accidents initiated by the external event. The quantity $f_k(z)$ can be expressed as

$$f_k(z) = \iint_{\mathcal{D}} \cdots \int f\left[\bigcup_{j=1}^{J}\{S_j(\underset{\sim}{y}), k(z)\}\right] h(\underset{\sim}{x})\, d\underset{\sim}{x} \qquad (10\text{-}1)$$

where

$h(\underset{\sim}{x})\ d\underset{\sim}{x}$ = frequency of occurrence of the external event with hazard intensity represented by the parameter values between $\underset{\sim}{x}$ and $\underset{\sim}{x} + d\underset{\sim}{x}$. Note that $\underset{\sim}{x}$ is a vector whose components represent different variables associated with the hazard, and the integration is carried over the entire domain of $\underset{\sim}{x}, \mathscr{D}$.

$\underset{\sim}{y}$ = vector of responses at a component location (structure, piping system, or equipment). Note that the responses are functions of hazard-intensity variables $\underset{\sim}{x}$; that is, $\underset{\sim}{y} = G(\underset{\sim}{x}, \underset{\sim}{\epsilon})$, where $\underset{\sim}{\epsilon}$ denotes the uncertainties in the response analysis.

$S_j(\underset{\sim}{y})$ = accident sequence j; a minimal cut set of components (1 to m_j)

= $\{1 \cap 2 \cap \cdots \cap m_j\}$.

$k(z)$ = damage level z for consequence type k.

k = consequence type $k = 1,\ldots,K$ (e.g., early fatalities, latent-cancer fatalities, and property damage).

$\cup$ = symbol for the union of events.

$\cap$ = symbol for the intersection of events.

In Equation 10-1, the term

$$f\left[\bigcup_{j=1}^{J} \{S_j(\underset{\sim}{y}), k(z)\}\right]$$

is the frequency of occurrence of any one of the j sequences resulting in damage level z of consequence type k. For a particular sequence j, the frequency of occurrence, $f[S_j(\underset{\sim}{y})]$, is calculated as the joint frequency of failure of components $1,2,\ldots,m_j$ in a single occurrence of the external event; it is a function of component fragility and response $\underset{\sim}{y}$. The conditional frequency of exceeding damage level z of consequence type k given the accident sequence j is $f_{k|S_j,\underset{\sim}{x}}(z)$. Note that this is a function of $\underset{\sim}{x}$, the hazard intensity. The effect of the external event on the environment is considered in this evaluation.

Since different accident sequences may involve some common components, the sequences are interdependent. The evaluation of the union of sequences in Equation 10-1 should take into account any correlations between component failures. These correlations arise from common structural models, single hazard input, and similar equipment (e.g., common vendor and identical mounting). There may also be some environmental dependence between component-failure events; for example, the collapse of a wall may damage a number of components simultaneously.

For the purposes of comparison with other external and internal events, and for merging with the risk analysis of internal events, the frequency of core melt from the external event and the frequencies of occurrence of different release categories can be calculated as shown below. The frequency of occurrence of core melt from an externally initiated accident, f_c, is expressed as

$$f_c = \iint \cdots \int f\left\{\bigcup_{j=1}^{J_c} S_{j,c}(\underset{\sim}{y})\right\} h(\underset{\sim}{x})\ d\underset{\sim}{x} \qquad (10\text{-}2)$$

where $S_{j,c}(\underset{\sim}{y})$ is core-melt sequence j_c ($j_c = 1,\ldots,J_c$):

$$S_{j,c}(y) = \left\{1 \cap 2 \cap \cdots \cap m_{j,c}\right\}$$

The frequency of occurrence of a release category, $\gamma = 1,\ldots,\Gamma$, from an accident initiated by an external event, f_γ, is expressed as

$$f_\gamma = \iint \cdots \int f\left\{\bigcup_{j=1}^{J_\gamma} S_j(\underset{\sim}{y})\right\} h(\underset{\sim}{x})\ d\underset{\sim}{x} \qquad (10\text{-}3)$$

where S_j is the accident sequence contributing to release category γ.

The total frequency $f_k^E(z)$ of exceeding damage level z of consequence type k resulting from all external events is approximately

$$f_k^E(z) \approx \sum f_k(z) \qquad (10\text{-}4)$$

Equation 10-4 is based on the assumption that the external events are statistically independent and that the frequencies of the simultaneous occurrence of two or more external events are small. However, the PRA analyst should study the possible dependence between external events. Note that the above formulation accommodates dependence by describing the hazard intensity and response in terms of vectors. This facilitates the treatment of multiple secondary events arising from a single external event. For example, a severe storm can produce concurrent flooding, high winds and associated missiles, and dam overtopping. The effects of some dependences have been considered in past PRA studies. For example, seismically induced dam failures and pipeline failures are considered in seismic risk analysis; in the hazard modeling, certain ambient conditions (i.e., waves, snowpack, etc.) are included. Although two external events may not simultaneously exert stress on a specific nuclear plant component (structure, piping, and equipment), they may affect different components in the same accident sequence (i.e., an earthquake may fail the reactor components, whereas flooding may damage the service-water pumps in the crib house). Also, the effect of one external event may be to induce a radionuclide release as a result of a reactor accident, whereas the other external event may modify the parameters of the consequence model.

The sections that follow present methods for evaluating different elements of Equations 10-1 through 10-3.

10.3.3 HAZARD ANALYSIS

A hazard analysis estimates the frequency of occurrence for different intensities of an external event, called "hazard intensities." It may be performed by developing a phenomenological model of the event, with parameter values estimated from available data and expert opinion. Alternatively, the hazard analysis may consist of extrapolating historical data, if appropriate. It should be noted that a hazard event can be described adequately only by a multitude of variables. For example, tornado hazard is described by the rate of occurrence, tornado path width, path length, translational wind speed, tangential wind speed and vertical velocity, and the number and types of objects that are potential missiles. One or more of these variables may be probabilistically dependent on other variables. Although the hazard model may be described in terms of some of these variables, the output of the analysis is generally expressed in terms of a limited number (typically, one) of variables. The tornado hazard may, for example, be characterized by site wind speeds (i.e., frequencies of exceeding different site wind speeds). The other variables that are necessary for a "complete" description of the hazard are to be considered in the response analysis and fragility evaluation. The particular variables(s) chosen to present the results of the hazard analysis may also depend on the plant-system and sequence analysis.

Typically, the output of hazard analysis is a hazard curve of exceedence frequency versus hazard intensity. Since there may be a great deal of uncertainty in the parameter values and in the mathematical model of the hazard, it is important to represent the effects of uncertainty (see Section 10.3.4.6) through a family of hazard curves. Each curve is plotted for a postulated set of parameter values and a selected hazard model. A probability value, P_i, is assigned to each curve. An example of a family of hazard curves is shown in Figure 10-2. For a discrete event, the result of the hazard analysis would be a probability distribution of the frequency of occurrence. An example of this type of event is the turbine-generated missile. The 95-percent probability interval of the annual frequency of turbine-missile generation could be reported as 10^{-5} to 10^{-3}.

The seismic hazard analysis described in Chapter 11 is a good example of the phenomenological approach. Chapter 11 also describes the analyses for fires and floods, which have emphasized the analysis of historical data. A detailed description of an aircraft-hazard analysis is available in a report published by a committee of the American Society of Civil Engineers (1980), which also lists some significant references on the topic. Hazard analyses for extreme winds and tornadoes have been described by Abbey (1976), Fujita (1971), Wen and Chu (1973), Garson et al. (1975), Wen (1976), Twisdale et al. (1978), and Simiu et al. (1979). Hazard analyses for accidents at industrial or military facilities, pipeline accidents, and transportation accidents are described in guidelines published by the American Nuclear Society (1978), which also contain an extensive bibliography, as well as reports by Cave and Kazarians (1978) and Eichler and Napadensky (1978). Details on the analysis of turbine-missile hazards are available in reports published by the American Society of Civil Engineers (1980), Bush (1973, 1977), and the Electric Power Research Institute (1981). Hazards from the onsite storage of chemicals are evaluated on the basis of quantity, distance from the control room, and the detection capabilities of the control room.

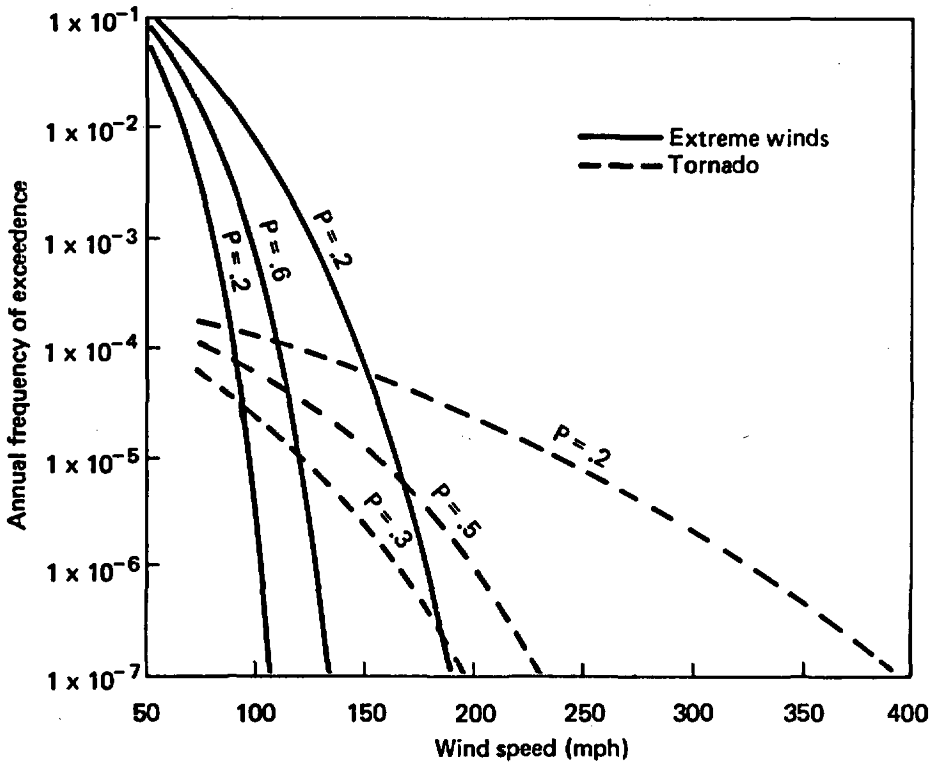


Figure 10-2. Family of hazard curves.

If the results of the hazard analysis and bounding evaluation of consequences show that the external event is not a significant contributor to risk, the PRA analyst may decide not to perform a more-detailed risk assessment. For example, if the impact of an aircraft or a turbine missile on any one of the safety-related structures has the potential for inducing a core melt but its frequency is much lower than that of the core-melt frequencies from other events, the event is ignored in further analysis. If the mean frequency for the rupture of a natural-gas pipeline near the plant is 10^{-7} per year, the effects of such a rupture (i.e., overpressure, missiles, and fire) are evaluated by using upper-bound assumptions. If the plant systems and structures are judged capable of withstanding these effects, the frequency of core melt from the pipeline rupture is assumed to be negligible.

Hazard analysis for fire and internal flooding may become intractable if all potential sources (i.e., locations) of hazard are to be considered. Procedures for screening significant source locations are discussed in Chapter 11.

10.3.4 ANALYSIS OF PLANT SYSTEM AND STRUCTURE RESPONSES

The purpose of this analysis is to translate the hazard input $\underset{\sim}{x}$ into the responses $\underset{\sim}{y}$ acting on a component. This generally involves an analysis of the structures, piping systems, and equipment. The hazard input could be a set of earthquake time histories, wind forces at selected elevations of the structures, the impact of an aircraft at a chosen location, or an incident pressure pulse due to a transportation accident. For each hazard intensity, the output of the response analysis would be a frequency distribution of the responses, such as spectral acceleration, peak moment, force, and deflection. The specific responses that are calculated depend on the failure modes of components. In the response analysis, any correlation between component responses resulting from the same hazard may be identified. When plant-design analysis information is considered appropriate, it may be used to estimate the structural responses for some external events. This circumvents the need for a detailed response analysis (Commonwealth Edison Company, 1981).

Some external events may not induce stresses in structures or components (e.g., a release of chemicals stored at the site and fire in a compartment). The response of the plant system (i.e., component and operator) needs to be considered in developing the accident sequences for such events. The propagation of fire and gases (e.g., smoke and chemicals) inside the plant determines which components and systems are affected.

10.3.5 EVALUATION OF COMPONENT FRAGILITY AND VULNERABILITY

The fragility or vulnerability of a component is defined as the conditional frequency of its failure given a value of the response parameter. For example, assume the wind hazard is characterized by the wind speed and let the wind speed be V_0. The response (e.g., force) due to this wind speed at a component location is R_0. The component's capacity to withstand the wind force is a random variable, C. The fragility of the component is calculated as

$$f = \text{frequency} \left\{ C < R_0 \right\} \tag{10-5}$$

If the component capacity is modeled as a lognormally distributed random variable with median $\check{C}$ and logarithmic standard deviation β, then f is calculated as

$$f = \Phi \left[\frac{\ln(\check{C}/R_0)}{\beta} \right] \tag{10-6}$$

where $\Phi(\cdot)$ is the standard Gaussian cumulative distribution function.

The fragility is estimated from the actual capacity of the component in any given failure mode. However, in estimating the capacity, uncertainties arise from several sources: an insufficient understanding of structural material properties and failure modes, errors in the calculated response due to approximations in modeling, and the use of generic data and engineering judgment in lieu of plant-specific data. Component fragility is therefore represented by a family of fragility curves. Each curve is developed on the basis of an assumed set of parameter values and failure modes. A probability q_i is assigned to this curve. The development of seismic fragility curves is explained in Chapter 11.

In some applications, the fragility parameter is taken as the hazard intensity; the capacity of the component, derived from design criteria and test data, is expressed in terms of the hazard intensity, using design-analysis information. The example given below shows how fragility curves are developed for a structure subject to wind loading. Let the design wind speed be V_d (for example, it could be 80 mph, corresponding to a mean recurrence interval of 100 years, at a reference height of 33 feet). The structure capacity C (mph) for wind loading can be expressed as

$$C = V_d F_W F_S \tag{10-7}$$

Here F_W is the safety factor relating the design wind pressure to the actual wind pressure on the structure; it is a function of terrain (exposure), peak pressure fluctuations, and gust response. The safety factor F_W is expressed as

$$F_W = \check{F}_W \varepsilon_{W,R} \varepsilon_{W,U} \tag{10-8}$$

where $\check{F}_W$ is the median safety factor, $\varepsilon_{W,R}$ is a random variable reflecting the inherent randomness in the wind pressure, and $\varepsilon_{W,U}$ is a random variable reflecting the uncertainty in the calculation of $\check{F}_W$. Both $\varepsilon_{W,R}$ and $\varepsilon_{W,U}$ are assumed to be lognormally distributed with logarithmic standard deviations $\beta_{W,R}$ and $\beta_{W,U}$, respectively. The values of $\beta_{W,R}$ and $\beta_{W,U}$ are taken as 0.20 and 0.30, respectively. The other quantity in Equation 10-7, F_S, is the safety factor relating the actual capacity of the structure to the calculated capacity. It is a function of the allowable stresses, the complete spectrum of load conditions for which the structure is designed, material strength variations, and approximations in structure modeling. The median value of $\check{F}_S$ is estimated as 1.5; the values of $\beta_{S,R}$ and $\beta_{S,U}$ are taken as 0.15 and 0.35, respectively.

Note that F_W and F_S are expressed as ratios of wind speeds. With these values, the median $\check{C}$ and the logarithmic standard deviations $\beta_{C,R}$ and $\beta_{C,U}$ of C are calculated as

$$\check{C} = 1.5V_d \tag{10-9}$$

$$\beta_{C,R} \approx (0.20^2 + 0.25^2)^{1/2} = 0.25 \tag{10-10}$$

$$\beta_{C,U} \approx (0.35^2 + 0.15^2)^{1/2} = 0.38 \tag{10-11}$$

Using Equations 10-9 through 10-11 and the lognormal-distribution assumption, the fragility of the structure, f', at a wind speed V, at any nonexceedence probability level Q can be derived by using the formulation given by Kennedy et al. (1980):

$$f' = \Phi\left[\frac{\ln(V/\check{C}) + \beta_{C,U}\Phi^{-1}(Q)}{\beta_{C,R}}\right] \tag{10-12}$$

where $Q = Pr[f < f'|V]$ is the probability that the true conditional failure frequency f is less than f' given a wind speed V and where $\Phi^{-1}(\cdot)$ is the inverse of the standard Gaussian cumulative distribution function. Note that Q is the sum of the probabilities assigned to all the fragilities less than f'. By this formulation, both the inherent randomness and the uncertainty are explicitly represented. Figure 10-3 shows a family of fragility curves for the structure. Such fragility curves are developed for different components whose failures are identified as either initiating an accident or contributing to any significant accident sequence that would result in a core melt or the release of radionuclides.

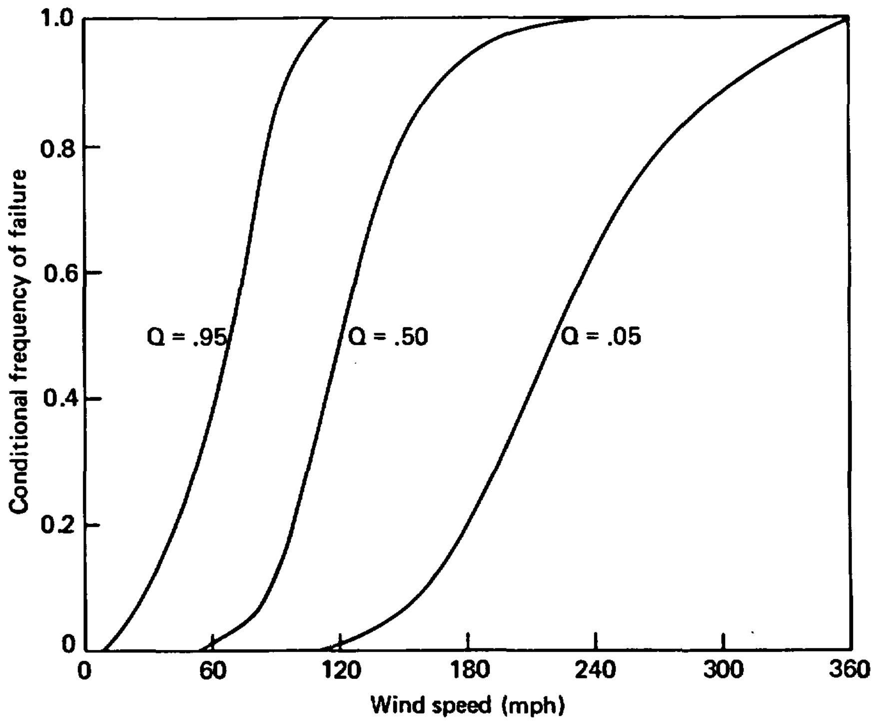


Figure 10-3. Fragility curves for wind loading.

For some external events, component fragility can be taken as 1.0 if the hazard intensity reaches a specific value (e.g., the inundation of equipment by flooding and turbine-missile impact on a vital component). The fragilities of individual components along with the information on correlation of responses and of capacities between components can be used to calculate the conditional frequencies of accident sequences consisting of a number of components. For some external events (e.g., missiles, internal flooding, and fire), the fragilities of individual components may not be meaningful; the conditional frequencies of accident sequences are directly calculated (e.g., by counting how many times a turbine-missile trajectory passed through a sequence of components).

Before the individual component fragilities can be combined in the plant-system logic, which is discussed in Section 10.3.6, it is necessary to evaluate the degree of dependence to be assumed among the failure frequencies in sets of multiple components (i.e., minimal cut sets). An incorrect assumption that the failures occur independently can lead to optimistic predictions if the multiple components appear within the same minimal cut set or to conservative predictions if they are not in the same cut set. Specific examples of the dependence or correlation of component fragilities are given in Chapter 11.

10.3.6 ANALYSIS OF PLANT SYSTEMS AND EVENT SEQUENCES

The analysis of plant systems and event sequences consists of developing event trees and fault trees in which the initiating event can be the external hazard itself or a transient or LOCA initiating event induced or caused by the external event. Various failure sequences that lead to core melt, containment failure, and a specific release category are identified. The component fragilities are then used to compute the frequencies of the various event sequences. These calculated frequencies are conditional on the specified hazard intensity. The unconditional frequency of core melt or of radionuclide release for a given release category is obtained by integrating over the entire range of hazard intensities. The consequence analysis can be carried out separately for each external event when appropriate. The output of the external event analysis would then be curves of the frequencies of damage (i.e., acute fatalities, latent-cancer fatalities, or property damage) at different nonexceedence probabilities.

If the external event analysis is merged with the internal event analysis at the stage of event-tree development, the analyst should provide the information on the initiating events for each range of hazard intensity, necessary modifications to the event trees, complete fault trees, changes to the containment event tree, and differences in the consequence-analysis results, along with the hazard curves and component-fragility curves. The component-failure dependences resulting from a common hazard intensity and similar equipment should be explicitly represented in the fault trees.

If the external event analysis is merged with the internal event analysis at the consequence-analysis stage, the analyst should provide the probability distribution of the frequency of release for each release category. This probability distribution can be calculated by event-tree and fault-tree methods. In some instances, simplified plant-level fault trees are formulated for the core melt; the type of core melt (i.e., the plant state) is decided on the functioning of fan coolers and containment sprays. The plant states are aggregated with the containment states in order to determine the release category to which they belong. A Boolean expression in terms of component failures is derived for each release category. Component fragilities are used in this expression to compute the plant-level fragility family for each release category. This family, when integrated over the hazard curves, gives the probability distribution of the frequency of release for each release category. An example of these probability distributions is shown in Figure 10-4.

If the analyst decides to keep the analysis of an external event totally separate from the analysis of internal and other external events, the probability distributions of frequency of release categories are input into the consequence-analysis model developed for the external event. The consequence-analysis modeling may depend on the external event. For example, a large earthquake or an external flood may disrupt the communications network and damage the evacuation routes (the evacuation time was increased in the Diablo Canyon Seismic Risk Study (Pacific Gas & Electric Company, 1977) to account for the effect of large seismic events on roads, bridges, structures, and communications); and extreme winds may carry radioactive materials to more distant locations. For this level of external event analysis, the output is the final risk curve plotted for a specified nonexceedence probability level. This may be compared with the risk curves from other internal or external events to judge the relative risk significance of the events under study.

The interfaces between the analysis of plant systems and event sequences and those aspects of PRA peculiar to external events--namely, the analyses of hazard and hazard response--can be seen from the following example. Suppose that a nuclear plant consists of two systems, 1 and 2, each of which protects against core damage in response to some arbitrary initiating event, denoted by X. For example, X might represent a transient event and the two systems might be the auxiliary feedwater system and the collection of components and operator actions required for "feed and bleed" cooling, respectively. Further, suppose that system 1 and system 2 each consists of two redundant subsystems, denoted by A and B for system 1 and C and D for system 2. The event tree and the list of minimal cut sets for this simple example are presented in Figure 10-5.

Accident sequences resulting from some external event can be viewed as "superimposed" on the event tree in Figure 10-5 in the following way: The external event, denoted in this example by E, can produce or contribute to an accident sequence either by causing the initiating event X to occur, or causing the failure of one or more subsystems, or a combination of these. Alternatively, accident sequence S_3, for example, can result from any combination of external and nonexternal causes that results in the failure of systems 1 and 2 and the occurrence of the initiating event.

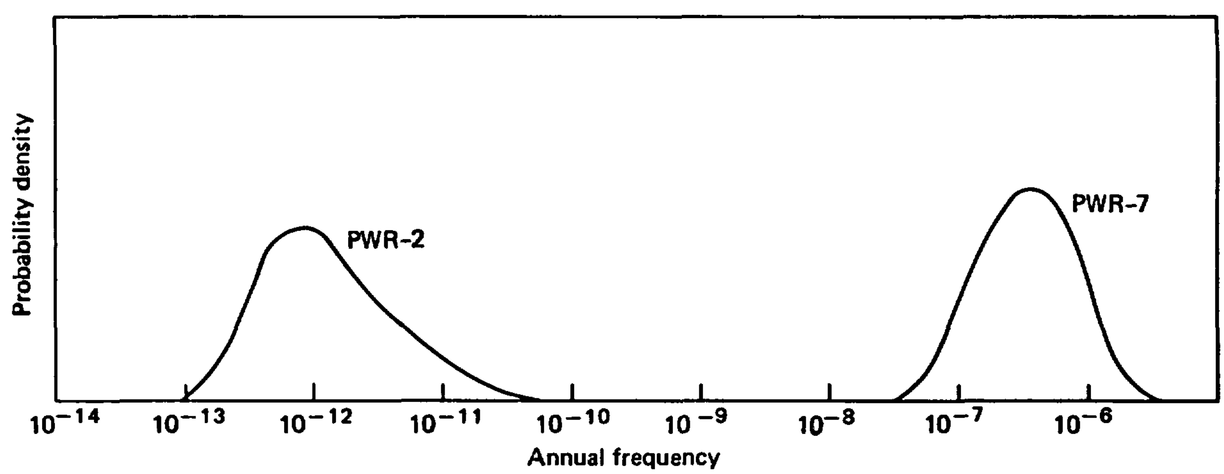
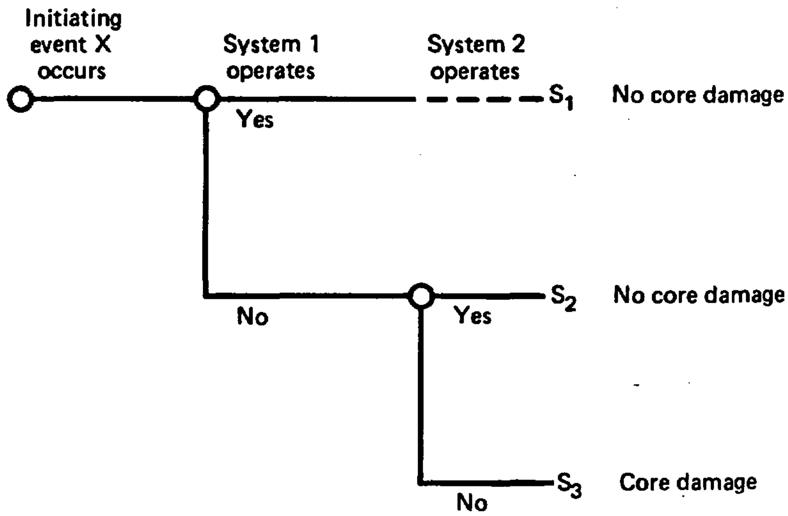


Figure 10-4. Release frequency from extreme-wind event for two release categories: PWR-2 and PWR-7.



System	Minimal cut sets
1	{A, B}
2	{C, D}

Figure 10-5. Example event tree indicating the responses of front-line systems 1 and 2 to initiating event X.

The interfaces between the event- and fault-tree methods, which are generic and applicable to all initiating events, and the analysis of hazard frequencies and responses, which are specific to external events, can be seen by structuring the accident sequences into the following sequence fragments:

1. The external event occurs.

2. Damage causally related to external event occurs.

3. The initiating event occurs.

4. Any additional failures not causally related to the external event occur.

5. Accident terminates in some damage state.

The tasks of hazard analysis and the analysis of the response of structures, systems, and components to the external events are then associated with sequence fragments 1 and 2, whereas sequence fragments 3, 4, and 5 are common to all accident sequences due to both internal and external causes.

As illustrated in Figure 10-6, there are 16 possible sequence fragments, referred to as external event damage states, that provide a complete representation of the possible effects of the external event on the four subsystems in our simple example. To the extent that the sequences in Figure 10-5 are a complete set, a complete representation of the accident sequences associated with external event E can be constructed by feeding each of the 16 sequence fragments in Figure 10-6 into the event tree in Figure 10-5, for a total of 48 sequences. (Note that, in this particular example, some of these 48 sequences would have zero frequency. For example, if external event damage state E_{16} occurs, it is impossible for sequences S_1 or S_2 to occur.)

Estimates of the frequency of core damage in the example can be made by applying Equation 10-2 in the modified form of

$$f_c = \int_{\mathcal{D}} \cdots \int f_{S,3}(\underset{\sim}{y})\ h(\underset{\sim}{x})\ d\underset{\sim}{x}$$

$$= \int_{\mathcal{D}} \cdots \int f\left[\bigcup_{\ell=1}^{16} \{E_\ell(y), S_3\}\right] h(\underset{\sim}{x})\ d\underset{\sim}{x} \qquad (10\text{-}13)$$

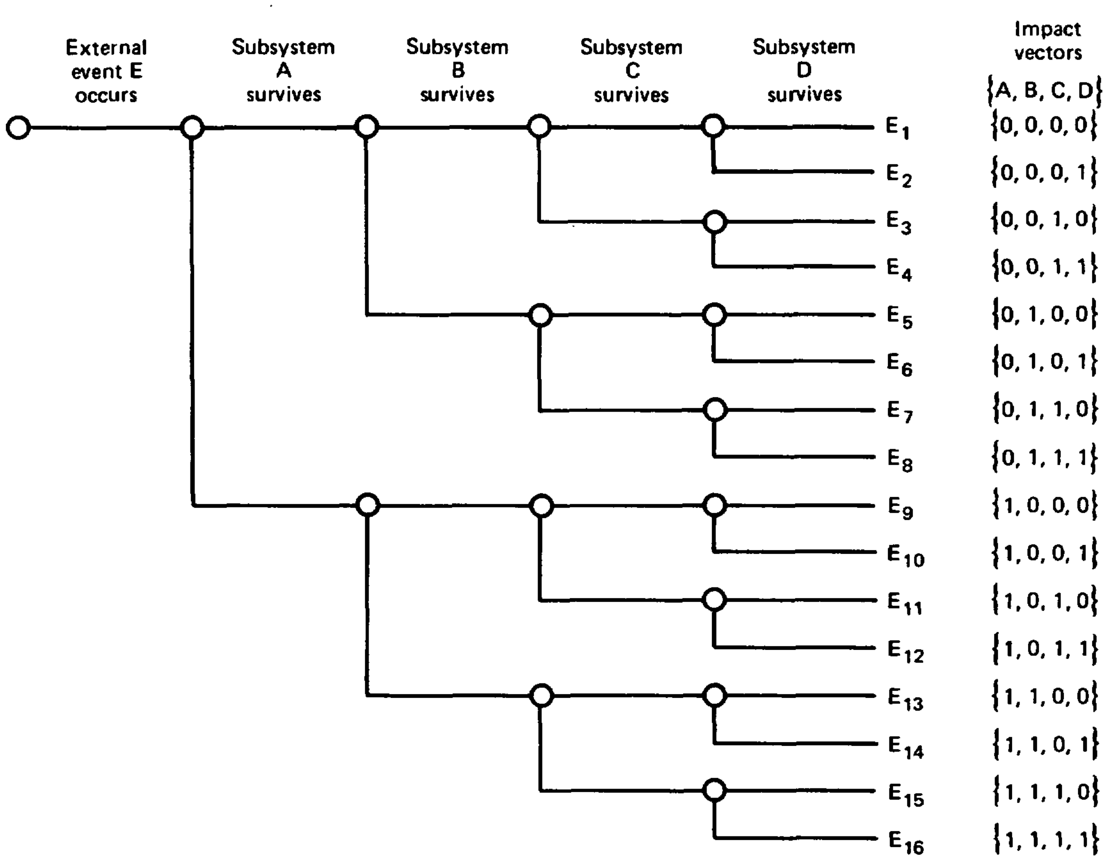


Figure 10-6. External event tree and associated impact vectors illustrating the possible damage states of four subsystems A, B, C, and D. In the heading of the tree "subsystem survives" means that the subsystem does not fail as a direct consequence of the external event. For the impact vectors, 0 indicates survival and 1 indicates a failure of the particular subsystem.

where $f[\cup_{\ell=1}^{16} \{E_\ell(\underset{\sim}{y}), S_3\}]$ is the frequency of occurrence of any one of the sequence fragments (or external event damage state) E_ℓ given response $\underset{\sim}{y}$ to the external event and resulting in the accident sequence S_3. An approximation for f_c is

$$f_c \approx \int_{\mathscr{D}} \cdots \int \sum_{\ell=1}^{16} f_{E,\ell}(\underset{\sim}{y}) \; f_{S,3|E,\ell} \; h(\underset{\sim}{x}) \; d\underset{\sim}{x} \qquad (10\text{-}14)$$

where $f_{E,\ell}(\underset{\sim}{y})$ is the frequency of sequence fragment E_ℓ given response $\underset{\sim}{y}$ to the external event and $f_{S,3|E,\ell}$ is the conditional frequency of accident sequence S_3 given external event damage state E_ℓ occurs.

Note that the assessment of the frequency of the external event damage state, $f_{E,\ell}(\underset{\sim}{y})$, involves a specific combination of failure and successes resulting from the external event. It is very important to recognize that dependences or correlations may preclude the synthesis of these frequencies as an independent combination of component fragilities.

The quantity $f_{S,3|E,\ell}$ represents the interface between the analysis of external events and the PRA analyses generic to all accident sequences. It can be estimated by using the basic event- and fault-tree methods described in Chapters 3 through 6 for all initiating events. It is extremely important, however, that the boundary conditions for quantifying the accident frequency reflect the impact of the external event. Those boundary conditions can be represented in the form of an "impact vector" (see Figure 10-6). In the quantification of the front-line event tree in Figure 10-5 for each external event damage state, E_ℓ, the impact vector denotes the subsystems that are failed as initial conditions in the quantification. For example, since external event damage state E_{16} results in the failure of minimal cut sets in both systems, it follows that $f_{S,3|E,16} = 1$.

An alternative representation that indicates the relationship between the external events and the generic aspects of PRA methods is a fault tree that is constructed for an entire accident sequence or a collection of sequences referred to as a plant-damage state or bin. Such a fault tree, constructed for the top event "core damage," is presented in Figure 10-7 for the earlier example. The fault tree indicates that each of the subsystems A, B, C, and D has both external and nonexternal failure causes and that the frequency of the initiating event depends on the external event. The quantification of this tree would produce a result equivalent to Equation 10-13. As discussed in Chapter 11, some of the seismic risk analyses have been carried out with the use of fault-tree logic to model plant-damage states similar to that in Figure 10-7.

In practical applications of external event risk analysis it is neither feasible nor desirable to provide such a complete enumeration of the possible combinations of external event damage states and nonexternal failure causes as illustrated in the above example. Indeed, it was possible to give a "complete" representation only because the example includes as few as four subsystems and only one accident sequence. To address the large number of subsystems and accident sequences postulated for a complex nuclear plant system, it is necessary to make some approximations and simplifications. One such simplification is to reduce the number of external event damage

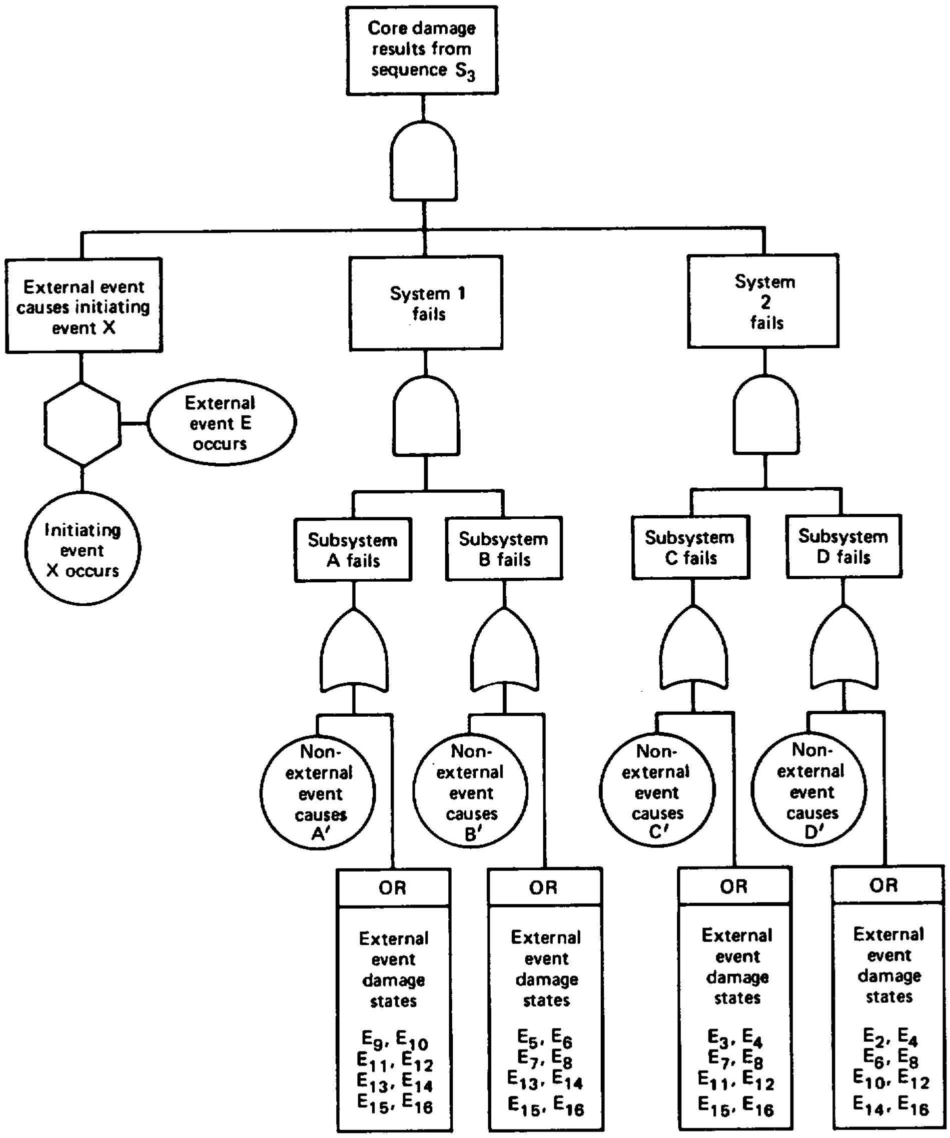


Figure 10-7. Fault tree for core damage due to external event.

states to a relatively small number in relation to the large number of cases that can be postulated. For example, suppose that in our example the external event in question is a fire at a specific location near the four subsystems, as shown in Figure 10-8. Of the 16 possible fire damage states that can be postulated, it should only be necessary to consider the following five, which are described in terms of their impact vectors:

$$E_1 = \{0,0,0,0\}; \quad E_9 = \{1,0,0,0\}; \quad E_{13} = \{1,1,0,0\}$$

$$E_{15} = \{1,1,1,0\}; \quad E_{16} = \{1,1,1,1\}$$

The remaining 11 damage states can be dismissed because the above are representative of the complete spectrum of states and would be expected to occur at a much greater frequency.

A second type of simplification is to specialize the logic of the event and fault trees to different discrete intervals of hazard or response intensity. For example, at high levels of intensity that approach or exceed the capacities of the plant components, it may not be necessary to consider nonexternal causes in the model.

A third approach to keeping the amount of data processing at a manageable level is to separately quantify the external event model, such as the event tree of Figure 10-6, and eliminate the accident-sequence fragments that make negligible risk contributions before quantifying the front-line event trees, such as that in Figure 10-5. This is made possible by the use of the impact vectors, which provide a measure of the damage of the external event in terms of subsystem-failure impact. This same impact vector is used to help model certain types of intersystem dependences, as described in Section 3.7. By comparing the frequencies and impact vectors of all the external event damage states, the total number of damage states can often be reduced by combining states with similar or symmetrical impact vectors and eliminating states that are negligible risk contributors before integrating the external event and nonexternal event portions of the event- and fault-tree logic.

10.4 TREATMENT OF UNCERTAINTY

There are many uncertainties in the analysis of external events; they arise from lack of data and analytical models. In the hazard analysis, the uncertainties to be considered are those in the frequency of occurrence of the hazard intensity, the characterization of the phenomenon (e.g., line source or point source for seismic events, path width and length models for a tornado, available sources of missiles for a tornado, and models for

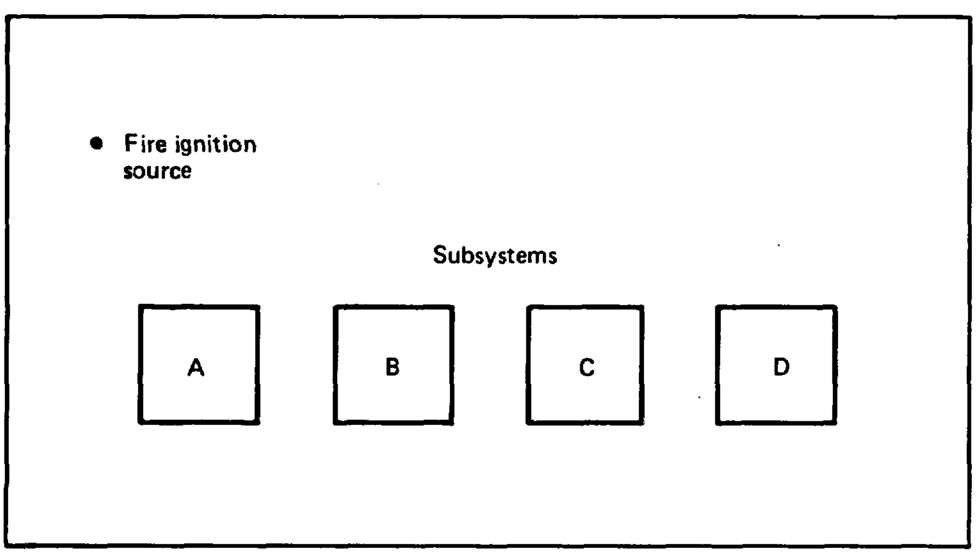


Figure 10-8. Hypothetical layout of subsystems in relation to fire.

explosive-vapor cloud transport), and the transmission of effects (e.g., overpressure, missiles, and ground acceleration) from the source to the site. In the component-fragility evaluation, uncertainties arise from an insufficient understanding of the properties and failure modes of structural materials, errors in the calculated response due to approximations in modeling, and the use of generic data and engineering judgment in the absence of plant-specific data.

At present, the quantification of uncertainty in different phases of the external event PRA is performed with a combination of limited analysis, sparse empirical data, and expert opinion. The uncertainty in hazard analysis is expressed by a family of hazard curves, each curve being drawn for a different nonexceedence-probability level. Similarly, the component fragility is expressed by a family of fragility curves, each curve being drawn for a specified nonexceedence-probability level.

One method of consistently propagating the uncertainties is to perform the risk analysis in two stages. In the first stage, the risk assessment is done by using the best-estimate hazard curve and the best-estimate fragility curve for each component. In the second stage, the risk assessment of the first stage is repeated many times, each time with a different set of hazard and component-fragility curves. These sets are sampled from the probability distributions of the hazard curves and the fragility curves reflecting their uncertainties. Since the uncertainties arise from an incomplete understanding of the phenomenon and from the use of simplified models, it is important to maintain correlations between the component fragilities in this sampling.

By performing this two-stage analysis a sufficient number of times, the probability distributions of core-melt frequency and the frequency of each release category can be obtained. A similar treatment of the uncertainties in the consequence analysis would yield the probability distribution of the exceedence of damage. Since this thorough treatment of uncertainty can become very expensive, the analyst should attempt to identify the dominant accident sequences and perform the uncertainty propagation for those sequences only.

If external events are analyzed with simplified plant-level fault trees, the uncertainties are propagated by assigning probability distributions for each component-failure frequency in the Boolean expressions. Usually, a family of curves for plant-level fragility for core melt and for each release category are obtained. Integration over the hazard-curve family then yields probability distributions for core-melt frequency and the frequency of each release category. Integration can be accomplished numerically by using discrete-probability-distribution arithmetic, the method of moments, Monte Carlo error propagation, response-surface analysis, or other statistical techniques discussed in Chapter 12.

Because of the large uncertainties present in the hazard analysis, component-fragility evaluation, plant systems, and the analyses of accident sequences, containment-failure modes, and consequences, it is important that the uncertainties be treated explicitly and consistently--and be propagated throughout the analysis in order to quantify the total uncertainty in the plant risk. Examples of available information on uncertainties are discussed in reference to seismic, fire, and flood risk analyses in Chapter 11.

10.5 INFORMATION AND PHYSICAL REQUIREMENTS

The plant design bases detailed in the safety analysis report should be reviewed for data on the site region and potentially hazardous activities in the vicinity of the plant. The analyst should ensure, however, that any conservative bias in the data is properly accounted for. This information is used with the models of external events to develop the frequencies of hazard intensities. The design criteria, applicable codes and standards, stress reports, material test data, design reports, location of plant safety systems and structures, dimensions of structural members, as well as reports on qualification and preservice tests and on periodic in-service inspection should be reviewed in order to develop the fragilities for components and systems.

10.6 DOCUMENTATION

The PRA report should contain a list of all external events that are identified as potential hazards, the screening criteria, and a table listing all excluded external events and giving the applicable screening criteria.

The report should contain a detailed description of the hazard analysis for each selected external event. The development of component fragilities, initiating events, event and fault trees, and containment event trees should be included. If the analysis is carried out independently of other external and internal events, the report should include the probability distributions of core-melt frequency, the frequencies of various release categories, and risk curves.

10.7 DISPLAY OF FINAL RESULTS

The results of the external event analysis described in this chapter will be the following:

1. The identification of external events appropriate to the site and plant.

2. The selection of the events for which a detailed risk assessment is done.

3. Hazard analysis, component fragilities, modifications to the event and fault trees and containment event trees, and modifications to the consequence model as input to the analyses described in Chapters 3 and 9.

4. Probability distributions of core-melt frequency, the frequencies of various release categories, and risk curves, if appropriate.

10.8 ASSURANCE OF TECHNICAL QUALITY

The provisions described in Chapter 2 for the assurance of technical quality are applicable to the external event analysis described in this chapter. The key elements are documentation, peer review of methods and data, and documentation of the parameters elicited from expert opinion.

NOMENCLATURE

C	capacity of a component
$\check{C}$	median capacity of a component
E_ℓ	a discrete level of damage described in terms of subsystems failed because of an external hazard (external event damage state)
F_S	safety factor relating the actual capacity of the structure to the calculated capacity
F_W	safety factor relating the design wind pressure to the actual wind pressure on the structure
$\check{F}_W$	median value of F_W
f	conditional failure frequency of a component; fragility
f'	fragility at a nonexceedence-probability level Q
f_c	frequency of core melt
f_γ	frequency of release by release category γ
$f_k(z)$	frequency of exceeding damage level z of consequence type k
$f_{k\|S,j}(z)$	conditional frequency of exceeding damage level z of consequence type k given accident sequence S_j
$f_k^E(z)$	total frequency of exceeding damage level z of consequence type k resulting from all external events
$f_{E,\ell}(\underset{\sim}{y})$	frequency of external event damage state E_ℓ
$f_{S,j}(\underset{\sim}{y})$	frequency of accident sequence S_j
$f_{S,i\|E,\ell}$	frequency of accident sequence S_i given damage state E_ℓ
$h(\underset{\sim}{x})\ d\underset{\sim}{x}$	frequency of occurrence of the external event with hazard intensity represented by parameter values between $\underset{\sim}{x}$ and $\underset{\sim}{x} + d\underset{\sim}{x}$
J_γ	number of accident sequences contributing to release category γ
k	consequence type $k = 1,\ldots,K$ (e.g., early fatalities, latent-cancer fatalities, and property damage)
m_j	total number of components in an accident sequence j
$m_{j,c}$	total number of components in a core-melt sequence

P, Q	nonexceedence-probability levels
p_i, q_i	probability assigned to hazard or fragility curve
R_0	response at a component location due to wind speed V_0
S_j	accident sequence $j = 1,\ldots,J$
$S_{j,c}$	core melt sequence $j_c = 1,\ldots,J_c$
V	wind speed
V_0	specific wind speed
V_d	design wind speed
$\underset{\sim}{x}$	a vector of hazard intensity parameters
$\underset{\sim}{y}$	a vector of responses
z	damage level of consequence type k
β	logarithmic standard deviation
$\beta_{C,R}$	logarithmic standard deviation reflecting the inherent randomness in the variable C
$\beta_{C,U}$	logarithmic standard deviation reflecting the uncertainty in C
γ	release category $\gamma = 1,\ldots,\Gamma$
$\mathcal{D}$	domain of $\underset{\sim}{x}$
$\varepsilon_{W,R}$	random variable reflecting the inherent randomness in the wind pressure
$\varepsilon_{W,U}$	random variable reflecting the uncertainty in the calculation of $\check{F}_W$
$\Phi(\bullet)$	standard Gaussian cumulative distribution function
$\Phi^{-1}(\bullet)$	inverse of the standard Gaussian cumulative distribution function

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